# Appendix C
# Health and Safety

## C.1 INTRODUCTION

Appendix C presents detailed information on the potential impacts and risks to humans associated with releases of radioactivity and hazardous chemicals from the HFBR during normal operations and from theoretical accidents. This information is intended to support the Public and Occupational Health and Safety assessments described in Sections 3.11 and 4.11 of this DEIS. Section C.2 provides information on radiological impacts from normal operations, Section C.3 provides information on hazardous chemical impacts from normal operations, Section C.4 provides information on human health and epidemiological studies, and Section C.5 provides information on theoretical facility accidents.

## C.2 RADIOLOGICAL IMPACTS TO HUMAN HEALTH DURING NORMAL OPERATIONS

This section provides background information on the nature of radiation (Section C.2.1), the methodology used to calculate radiological impacts (Section C.2.2), and the radiological releases resulting from normal operation of the HFBR (Section C.2.3).

### C.2.1 BACKGROUND

#### C.2.1.1 Nature of Radiation and Its Effects on Humans

**What is Radiation?** Humans are constantly exposed to radiation from the solar system and from the earth's rocks and soil. This radiation contributes to the natural background radiation that has always surrounded us. But there are also manufactured sources of radiation, such as medical and dental x-rays, household smoke detectors, exit signs, and materials released from nuclear and coal-fired power plants.

Ordinary matter is composed of atoms, and radiation comes from the disintegration of these tiny particles. Atoms are made up of even smaller particles called protons, neutrons, and electrons. The number and arrangement of these particles distinguishes one atom from another.

All atoms are elements. Elements are distinguished by the number of protons they contain. There are over 100 natural and man-made elements. Some of these elements, such as uranium, radium, plutonium, and thorium, share a very important quality: they are unstable. As they change into more stable forms, invisible waves of energy or particles, known as ionizing radiation, are released. Radioactive decay is the process of emitting this radiation.

Ionizing radiation refers to the fact that the energy emitted during radioactive decay can ionize, or electrically charge atoms by stripping or knocking electrons away from the nucleus of the atom that they surround. Ionizing radiation can cause a change in the chemical composition of many things, including living tissue and organs, which can affect the way they function.

The radiation that is emitted during the disintegration (decay) of a radioactive substance has different effects on people, depending on the kind of radiation (alpha and beta particles, gamma and x-rays, and neutrons) and the total amount of radiation energy absorbed by the body. The total energy absorbed per unit mass of tissue is called the "absorbed dose." The absorbed dose, when multiplied by certain quality factors and factors that take into account the different sensitivities of various tissues, is called the "effective dose equivalent," (EDE) or where the context is clear, simply "dose." The common unit of EDE is the "rem." A thousandth of a rem is a "millirem," abbreviated as "mrem."

Alpha particles, which each contain two neutrons and two protons, are the heaviest of these direct types of ionizing radiation, and can travel only several centimeters (a few inches) in the air. Alpha particles lose their energy almost as soon as they collide with anything. They can easily be stopped by a sheet of paper or the skin's surface.

Beta particles, which are identical to electrons, are much lighter than alpha particles. They can travel in the air for a distance of about 3 m (10 ft). Beta particles can pass through a sheet of paper but may be stopped by a thin sheet of aluminum foil or glass.

Gamma and x-rays, unlike alpha or beta particles, are waves of pure energy. Gamma radiation is very penetrating, and high energy gamma rays require a thick wall of concrete, lead, or steel to stop them.

The neutron is another particle that contributes to radiation exposure, both directly and indirectly. Indirect exposure is associated with the gamma rays and alpha particles that are emitted following neutron capture by the nucleus of an atom. A neutron has about one quarter the weight of an alpha particle. Neutrons are more penetrating than beta particles, but less than gamma rays.

The radioactivity of any material decreases with time. The time it takes a material to lose half of its original radioactivity is its half-life. For example, a quantity of $I^{131}$, a material that has a half-life of approximately eight days, will lose half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Eventually, the radioactivity will essentially disappear. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years or more.

As a radioactive element undergoes radioactive decay, it often changes to an entirely different element, one that may or may not be radioactive. Eventually, a stable element is formed. This transformation may take place in several steps and is known as a decay chain. Radium, for example, is a naturally occurring radioactive element with a half-life of approximately 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of approximately 3.8 days. Radon decays to polonium and, through a series of steps, to bismuth, and ultimately to stable lead.

**Units of Radiation Measure:** Scientists and engineers use a variety of units to measure radiation. These different units can be used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or its effects using units of calories or degrees, amounts of radioactivity can be measured in Curies.

The Curie, named after the French scientists Marie and Pierre Curie, is defined as the quantity of radioactive material that decays at $3.7x10^{10}$ disintegrations (decays) per second. The rate of decay of 1 gram of radium is the basis of this unit of measure.

As mentioned previously, the total energy absorbed per unit mass of tissue is called an "absorbed dose". The "rad" is the unit of measurement for the physical absorption of radiation. Radiation gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

A "rem" is a measurement of a dose of radiation based on its biological effects. The rem is used to measure the effects of radiation on the body. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other type of radiation. This standard allows comparison of the biological effects of radionuclides that emit different types of radiation.

An individual may be exposed to ionizing radiation externally from a radioactive source outside of the body or internally from radioactive material inside the body. The external dose is different from the internal dose. An external dose is delivered only during the actual time of exposure to the external radiation source. An internal dose, however, continues to be delivered as long as the radioactive source is in the body, although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes will decrease the dose rate with the passage of time. For regulatory purposes, the dose from an intake of radioactive material into the body is calculated over 50 years following the intake.

The three types of doses calculated in this EIS include an external dose, an internal dose, and a combined external and internal dose. Each type of dose is discussed below.

*External Dose:* The external dose can arise from several different pathways. (A "pathway" is the route through which radiation is received.) The radiation causing the exposure is external to the body in all of these pathways. In this EIS, these pathways include exposure to a cloud of radiation passing over the receptor (a "receptor" could be a member of the general public, an involved worker, or a noninvolved worker), standing on ground that is contaminated with radioactivity, swimming in contaminated water, and boating in contaminated water. The appropriate measure of dose is the effective dose equivalent (EDE).

*Internal Dose:* The internal dose arises from a radiation source entering the human body through either ingestion of contaminated food and water, inhalation of contaminated air, or absorption of contaminated material through the skin. Typical pathways for internal exposure include ingestion of crops contaminated either by airborne radioactive material deposited on the crops or by irrigation of crops using contaminated water sources, ingestion of animal products from animals that ingested contaminated food, ingestion of contaminated water, inhalation of contaminated air, and absorption of contamination in the water through the skin during swimming or bathing. Unlike external exposures, once the radiation enters the body, it remains there for various periods of time, depending on radioactive decay and biological elimination rates. The unit of measure for internal doses is the "committed dose equivalent." It is the internal dose that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.

The various organs of the body have different susceptibilities to harm from radiation. A weighting factor for each major organ or tissue is used to take these different susceptibilities into account. The weighting factor for each organ or tissue is applied to the committed dose equivalent in each organ or tissue, and the values for each organ or tissue are summed. The result is a broad indicator of the risk to the health of an individual from radiation and is called the "committed effective dose equivalent" (CEDE). The concept of "committed effective dose equivalent" applies only to internal pathways.

*Combined External and Internal Dose:* The sum of the CEDE from internal pathways and the EDE from external pathways is also called the "total effective dose equivalent" (TEDE) in this EIS (note that in DOE Order 5400.5, this quantity is called the "effective dose equivalent").

The units used in this EIS for committed dose equivalent, effective dose equivalent, and committed effective dose equivalent to an individual are the "rem" and "mrem." The corresponding unit for the collective dose to a population (the sum of the doses to members of the population, or the product of the number of exposed individuals and their average dose) is the "person-rem" or "person-mrem."

**Sources of Radiation:** The average person in the United States receives a total of about 360 mrem/yr from all sources of radiation, both natural and manufactured (NAS 1990). The sources of radiation can be divided into six different categories: cosmic radiation, terrestrial radiation, internal radiation, consumer products, medical diagnosis and therapy, and other sources. Each category is discussed below.

Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles and the secondary particles and photons they create are cosmic radiation. For the BNL site, the cosmic radiation is 28 mrem/yr (BNL 1996a). The average annual dose to the people in the United States is about 27 mrem.

External terrestrial radiation is the radiation emitted from the radioactive materials in the earth's rocks and soils. The average annual dose from external terrestrial radiation is about 28 mrem.

Internal radiation is caused by the human body metabolizing natural radioactive material that has entered the body by inhalation, ingestion, or absorption. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity are the short-lived decay products of radon, which contribute about 200 mrem/yr. The average dose from other internal radionuclides is about 39 mrem/yr.

Consumer products also contain sources of ionizing radiation. In some products, like smoke detectors and airport x-ray machines, the radiation source is essential to the products' operation. In other products, such as television and tobacco, the radiation occurs incidentally to the product function. The average annual dose from these sources is about 10 mrem.

Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average annual exposure of 39 mrem. Nuclear medical procedures result in an average annual exposure of 14 mrem.

There are a few additional sources of radiation that contribute minor doses to individuals in the U.S. The dose from nuclear fuel cycle facilities, such as uranium mines, mills and fuel processing plants, nuclear power plants, and transportation routes, has been estimated to be less than 1 mrem/yr. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from DOE and NRC facilities, emissions from certain mineral extraction facilities, and transportation of radioactive materials contributes less than 1 mrem/yr to the average dose to an individual. Air travel contributes approximately 1 mrem/yr to the average dose.

The collective (or population) dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This total dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 1 mrem (0.001 rem), the collective dose is 1,000 persons x 0.001 rem = 1 person-rem. Alternatively, the same collective dose (1 person-rem) results from 500 people, each of whom received a dose of 2 mrem (500 persons x 2 mrem = 1 person-rem).

**Limits of Radiation Exposure:** The amount of radiation to which the public may be exposed is limited by Federal regulations. Under the *Clean Air Act* (CAA), the exposure to a member of the general public from DOE facility releases into the atmosphere is limited by the EPA to a dose of 10 mrem/yr in addition to the natural background and medical radiation normally received (40 CFR 61, Subpart H). DOE also limits the dose annually received from material released to the atmosphere to 10 mrem (DOE 1993a). The EPA and DOE also limit the annual dose to a member of the general public from radioactive releases to drinking water to 4 mrem, as required under the *Safe Drinking Water Act* (SDWA) (40 CFR 141, DOE 1993a). The annual dose from all radiation sources from a site is limited by the EPA to 25 mrem (40 CFR 190). The DOE annual limit of radiation dose to a member of the general public from any DOE facility is 100 mrem total, from all pathways (DOE 1993a). For people working in an occupation that involves radiation, DOE limits individual doses to 5 rem (5,000 mrem) in any one year (10 CFR 835). The HFBR and the entire BNL site operate well below these limits.

It is estimated that the average individual in the United States receives a dose of about 0.3 rem (300 mrem) per year from natural sources of radiation. For perspective, a modern chest x-ray delivers an approximate dose of 0.006 rem (6 mrem), while a diagnostic pelvis and hip x-ray delivers an approximate dose of 0.065 rem (65 mrem) (NCRP 1987). A person must receive an acute (short-term) dose of approximately 600 rem (600,000 mrem) before there is a high probability of near-term death (NAS 1990).

### C.2.1.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this EIS places significant emphasis on the consequences of exposure to radiation, even though the effects of radiation exposure are small under most of the circumstances evaluated in this EIS. This section explains the basic concepts used in the evaluation of radiation effects in order to provide the background for later discussion of impacts.

Radiation can cause a variety of adverse health effects in people. Perhaps the most significant adverse health consequence of exposure to environmental and occupational radiation is the development of cancer, which may result in a fatality. The term "latent cancer fatality" is used to indicate if a fatality due to cancer caused by exposure to radiation is expected to occur.

Health impacts from radiation exposure, whether from sources external or internal to the body, are identified as "somatic" (affecting the individual exposed) or "genetic" (affecting descendants of the exposed individual). Except for leukemia, which can have an induction period (the time between exposure to the carcinogen and cancer diagnosis) of as little as two to seven years, most cancers have an induction period of more than 20 years. If a human body were to be irradiated uniformly, the incidence of cancer would vary among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. However, such cancers also produce relatively low mortality rates because they tend to be amenable to medical treatment. Because of the readily available data for cancer mortality rates and the relative scarcity of future epidemiological studies, somatic effects leading to latent cancer fatalities rather than cancer incidence are presented in this EIS. The number of latent cancer fatalities can be used to compare the risks among the various alternatives.

The National Research Council's Committee on the Biological Effects of Ionizing Radiations (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. One of these reports, *Health Effects of Exposure to Low Levels of Ionizing Radiation BEIR V*, published in 1990, provides the most current estimates for excess mortality from leukemia and cancers other than leukemia expected to result from exposure to ionizing radiation. The BEIR V report updates the models and risk estimates provided in the earlier report of the BEIR III Committee, *The Effects of Populations of Exposure to Low Levels of Ionizing Radiation*, published in 1980. BEIR V models were developed for application to the U.S. population.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiological data including the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy patients (breast cancer), New York postpartum mastitis patients (breast cancer), Israel Tinea Capitis patients (thyroid cancer), and Rochester thymus patients (thyroid cancer). Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry with an assumed Relative Biological Effectiveness of 20 for neutrons and were restricted to doses of less than 400 rads. Estimates of risks of latent cancer fatalities other than leukemia were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

**Risk Estimates for Doses Received During an Accident:** The BEIR V includes risk estimates for a single exposure of 10 rem to a population of 100,000 people (1.0 x $10^6$ person-rem). In this case, fatality estimates for leukemia, breast cancer, respiratory cancer, digestive cancer, and other cancers are given for both sexes and nine age-at-exposure groups. These estimates, based on the linear model, are calculated in terms of excess latent cancer fatalities per million person-rem and are summarized in Table C.2-1.

***Table C.2-1. Lifetime Risks per 100,000 Persons Exposed to a Single Exposure of 10 Rem***

| | Type of Latent Cancer Fatality | | |
|---|---|---|---|
| **Gender** | **Leukemia[a]** | **Cancers Other Than Leukemia** | **Total Cancers** |
| Male | 220 | 660 | 880 |
| Female | 160 | 730 | 890 |
| **Average** | **190** | **695** | **885[b]** |

[a] These are the linear estimates, and are double the linear-quadratic estimates provided in BEIR V for leukemia at low doses and dose-rates.

[b] This value has been rounded up to 1,000 excess latent cancer fatalities per million person-rem.

Source: NAS 1990.

The average risk estimate from all ages and both sexes is 885 excess latent cancer fatalities per million person-rem. This value has been conservatively rounded up to 1,000 excess latent cancer fatalities per million person-rem. Section C.5.1.2 contains additional discussions on the use of accident risk estimators for this EIS.

Although values for other health effects are not presented in this EIS, the risk estimators for non-fatal cancers and for genetic disorders to future generations are estimated to be approximately 200 and 260 per million person-rem, respectively. These values are based on information presented in the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP Publication 60) and are seen to be 20 percent and 26 percent, respectively, of the latent cancer fatality estimator. Thus, for example, if the number of excess latent cancer fatalities is projected to be "X," the number of excess genetic disorders would be 0.26 times "X."

**Risk Estimates for Doses Received During Normal Operation:** The BEIR V Committee found that a linear model for estimating health effects based on radiation exposure fit the data for all cancers except leukemia, and this linear model could be extrapolated to low doses. For leukemia caused by low doses and dose rates, the BEIR V Committee found that the health effects would be reduced by a factor of two from what would be predicted if using the linear model. For cancers other than leukemia, the BEIR V Committee recommended reducing the linear estimates by a factor between 2 and 10 for doses received at low dose rates (20 rem total). For this EIS, a risk reduction factor of 2 was adopted based on the fact that DOE, NRC, and EPA all use a risk reduction factor of 2.

Based on the above discussion, the resulting risk estimator would be equal to one-half the value observed for accident situations or approximately 500 excess latent cancer fatalities per million person-rem (0.0005 excess latent cancer fatalities per person-rem). This is the risk value used in this EIS to calculate latent cancer fatalities to the general public during normal operations. For workers, a value of 400 excess latent cancer fatalities per million person-rem (0.0004 excess latent cancer fatalities per person-rem) is used in this EIS. This lower value reflects the absence of children in the workforce. Again, based on information provided in ICRP Publication 60, the health risk estimators for non-fatal cancers and genetic disorders among the public are 20 and 26 percent, respectively, of the latent cancer fatality risk estimator. For workers, they are both 20 percent of the latent cancer fatality risk estimator. For this EIS, only latent cancer fatalities are presented.

The risk estimators may be applied to calculate the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to natural background radiation (0.3 rem/yr), 15 latent cancer fatalities per year of exposure would be inferred to be caused by the radiation (100,000 persons x 0.3 rem/yr x 0.0005 latent cancer fatalities per person-rem = 15 latent cancer fatalities per year).

Sometimes, calculations of the number of excess latent cancer fatalities associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1.0. This occurs because the latent cancer fatality determination is a statistical estimate. For example, if a population of 100,000 were exposed as above, but to a total dose of only 0.001 rem, the collective dose would be 100 person-rem, and the corresponding estimated number of latent cancer fatalities would be 0.05 (100,000 persons x 0.001 rem x 0.0005 latent cancer fatalities/person-rem = 0.05 latent cancer fatalities).

For latent cancer fatalities less than 1.0, the estimated 0.05 latent cancer fatalities are interpreted as a probability. That is, 0.05 is the *average* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, no one (zero people) would incur a latent cancer fatality from the 0.001 rem dose each member would have received. In a small fraction of the groups, one latent cancer fatality would result; in exceptionally few groups, two or more latent cancer fatalities would occur. The *average* number of deaths over all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 0.25). The most likely outcome is zero latent cancer fatalities.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of latent cancer fatalities" corresponding to a single individual's exposure over a (presumed) 72-year lifetime to 0.3 rem/yr is the following: 1 person x 0.3 rem/year x 72 years x 0.0005 latent cancer fatalities/person-rem = 0.011 latent cancer fatalities

Again, this should be interpreted in a statistical sense, that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1-percent chance that the individual might incur a latent cancer fatality caused by the exposure over his full lifetime. Presented another way, this method estimates that approximately 1.1 percent of the population might die of cancers induced by the radiation background.

## C.2.2 METHODOLOGIES FOR ESTIMATING RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS

### C.2.2.1 Data and Assumptions

The modeling used is primarily that which was used in the reference documents to estimate the type and amount of material released and the associated doses. These doses are converted to health effects using appropriate health risk estimators.

### C.2.2.2 Methodology for Dose Equivalent Calculations – Atmospheric Release Pathway

Doses from HFBR atmospheric releases were calculated using the Clean Air Act Assessment Package-1988 (CAP88-PC) dose model (EPA 1992). The CAP88-PC model uses a Gaussian plume equation to estimate the average dispersion of radionuclides released from elevated stacks or area sources. The program computes radionuclide concentrations in air, rates of deposition on ground surfaces and concentrations in food (where applicable) to arrive at a final value for projected dose at the specified distance from the release point to the location of interest. The program supplies both the calculated EDE to the exposed individual and the collective population dose within an 80 km (50 mi) radius of the emission sources. This model provides very conservative dose estimates in most cases (BNL 1999).

Input parameters used in the model include radionuclide type, emission rate in curies per year, and stack parameters such as height, diameter and exhaust velocity of the effluent. Site-specific weather data supplied by measurements from BNL's meteorological tower are used in the model. Data includes wind speed, direction, frequency, and temperature. A 10-year average data set for these meteorological parameters is used. Population data for the surrounding area is based on customer records of the Long Island Lighting Company (BNL 1999).

For this DEIS, the HFBR airborne radionuclide releases for the years 1988, 1995, and 1997 were used in the calculations because these were the years selected to represent the 60 MW operation Alternative, the 30 MW operation Alternative, and the No Action Alternative, respectively. The same population and site meteorology and 10-year wind averages were used in all the calculations.

## C.2.3 NORMAL OPERATIONS RADIOLOGICAL RELEASES

### C.2.3.1 Public Doses and Health Effects

Table C.2-2 provides the atmospheric releases attributable to the HFBR for the years 1988, 1995, and 1997. All of these releases were introduced into the atmosphere from a release height of 106 m (350 ft).

For 1988, all of the atmospherically released radionuclides shown in Table C.2-2 were calculated to contribute 0.069 person-rem to the population (Ports 1998a).

For 1995, all of the atmospherically released radionuclides shown in Table C.2-2 were calculated to contribute 0.035 person-rem to the population (Ports 1998d).

For 1997, all of the atmospherically released radionuclides shown in Table C.2-2 contributed 0.0098 person-rem to the population (BNL 1999).

***Table C.2-2. Atmospheric Radiological Releases Attributable to the HFBR for the Years 1988, 1995, and 1997***

| Radionuclide | Amount (Ci) |
|---|---|
| **1988** | |
| $H^{3}$ | 189 |
| $Br^{82}$ | 0.0025 |
| $I^{133}$ | 0.00026 |
| $I^{131}$ | 0.000057 |
| **1995** | |
| $H^{3}$ | 97.6 |
| $Ba^{128}$ | 0.0000098 |
| $Be^{7}$ | 0.00000098 |
| $Br^{77}$ | 0.0000023 |
| $Br^{82}$ | 0.0021 |
| $Co^{60}$ | 0.00000018 |
| $Cs^{137}$ | 0.00000003 |
| $I^{126}$ | 0.0000048 |
| $I^{131}$ | 0.0000014 |
| $K^{40}$ | 0.000065 |
| $Mn^{56}$ | 0.0000015 |
| $Xe^{133}$ | 0.0000083 |
| $Xe^{133m}$ | 0.00000056 |
| $Xe^{135}$ | 0.0000052 |
| **1997** | |
| $H^{3}$ | 27 |
| $Co^{60}$ | 0.000000057 |
| $Fe^{52}$ | 0.000000065 |
| $Rb^{84}$ | 0.000000088 |
| $Cs^{137}$ | 0.000000019 |

Source: BNL 1989, BNL 1996a, BNL 1999.

Table C.2-3 depicts the radiological impacts to the public for the 60 MW operation Alternative, the 30 MW operation Alternative, and the No Action Alternative (0 MW).

### C.2.3.2 Worker Doses and Health Effects

Worker doses were directly obtained from personnel dosimeters that are worn to monitor external exposures. Workers also participate in bioassay programs that monitor internal doses. Table C.2-4 provides data on HFBR worker doses and health effects for three years intended to represent 60 MW operation, 30 MW operation, and the 0 MW operation when the HFBR was not operating. The worker doses were projected by taking the average historical

***Table C.2-3. Annual Radiation Doses to the Public for Normal HFBR Operations***

| HFBR Characteristic | 60 MW | 30 MW | 0 MW |
|---|---|---|---|
| Total Dose (person-rem) | 0.069 | 0.035 | 0.0098 |
| Latent cancer fatalities | 0.000034 | 0.000017 | 0.0000049 |

Note:

1. For each power level, the total dose was calculated by the CAP88-PC model (EPA 1992) using a population of 5,053,187. The population input file for the CAP88-PC model was derived from customer records of LILCO (now LIPA). Note that, because of differences in population input file formats for the CAP88-PS model and the MACCS code (the computer code used to calculate accident radiological consequences (see SNL 1990a, SNL 1990b, SNL 1990c), a different offsite population (5,356,270) was used to calculate offsite accident doses. This population and associated offsite population distribution were calculated using SECPOP90 (Humphreys 1997). SECPOP90 is a computer program that provides population and economic data estimates for any location in the U.S. with the results available in MACCS site file format.
2. Latent cancer fatalities were calculated by using the dose-to-risk conversion factor for the public of 0.0005 latent cancer fatalities per person-rem.

Source: BNL 1989, BNL 1996a, BNL 1999, NAS 1990, Ports 1998a, Ports 1998d.

***Table C.2-4. HFBR Worker Doses and Health Effects for Three Operational Years***

| HFBR Characteristic | 60 MW | 30 MW | 0 MW |
|---|---|---|---|
| Total worker dose (person-rem) | 21.1 | 13.8 | 4.8 |
| Number of workers | 104 | 104 | 49 |
| Average worker dose (mrem) | 203 | 133 | 98 |
| Maximally exposed worker (mrem) | 870 | 634 | 513 |
| Latent cancer fatalities | 0.0084 | 0.0055 | 0.0019 |

Note: Latent cancer fatalities were calculated by using the dose-to-risk conversion factor for workers of 0.0004 latent cancer fatalities per person-rem.

Source: NAS 1990, Reciniello 1998.

worker doses for years that represent the reactor at the three different power levels and multiplying them by the expected workforces for the respective years.

# C.3 HAZARDOUS CHEMICAL IMPACTS TO HUMAN HEALTH

This section presents supporting information about the potential hazardous chemical impacts from the chemicals used and stored at HFBR. This section also provides background information related to those chemicals, and discusses the expected health impacts from potential releases of these chemicals.

## C.3.1 BACKGROUND

As a research facility, the HFBR does not have a standard set of chemicals or quantities of chemicals that are normally present within the complex. With the exception of standard industrial processes such as cooling water chemistry control, most of the chemicals are used and stored in a laboratory setting, where relatively small quantities of hazardous chemicals are used on a non-production basis. Changing research requirements used to, and would, necessitate the introduction of new substances as well as the removal or depletion of substances no longer required for experimental purposes. The hazards associated with each new chemical that would be introduced to the HFBR complex are expected to be evaluated on a case-by-case basis.

For this EIS, the following chemicals were selected based on their toxicity and their potential quantity at the HFBR: sulfuric acid, lithium arsenite, potassium hydroxide, cadmium nitrate, and lithium chromate. Table C.3-1 provides key characteristics concerning these chemicals which aids in understanding the hazards that they present.

The "Reference Dose" and "Reference Concentration" are set by EPA and represent exposure limits for long-term (chronic) exposure at low doses and concentrations, respectively, that can be considered safe from adverse non-cancer effects. The Permissible Exposure Limits (PELs) represent levels set by OSHA that are considered safe for 8-hour, daily-averaged exposures that will not cause non-cancer adverse effects. The cancer class identifies whether the chemical has been determined to be carcinogenic to humans.

## C.3.2 HAZARDOUS CHEMICAL RISK/EFFECTS

Specific analyses of the risks that the hazardous chemicals at the HFBR pose to the public and workers have not been performed. However, based on the nature and quantity of the chemicals at the HFBR, any hazardous chemical releases from the HFBR would be small and represent only a small percentage of the discharge levels allowed by Federal and State regulations. Thus, it is expected that there would be minimal public health impacts from hazardous chemical releases. Further, because discharges and emissions would vary little among the alternatives, public health effects would vary little as well.

HFBR operations may expose some workers to hazardous chemicals such as solvents, metals, and other chemicals that are carcinogenic.

***Table C.3-1. Key Characteristics of HFBR Chemicals***

| Chemical | Chemical Abstracts Service (CAS) No. | Reference Dose (oral) (mg/kg/day) | Reference Concentration (inhalation) ($mg/m^3$) | PEL ($mg/m^3$) | Threshold Planning Quantity (lbs) | Cancer Class[a] |
|---|---|---|---|---|---|---|
| Sulfuric acid | 7664-93-9 | 0.007 | 0.0245 | 1 | 1,000 | Not classified |
| Cadmium nitrate | 10325-94-7 | 0.0005[b] | Not established | 0.005[b] | Not listed | EPA Group B1[b] |
| Potassium hydroxide | 1310-58-3 | Not established | Not established | 2 | Not listed | Not classified |
| Lithium chromate | 14307-35-8 | 0.003[c] | 0.0001[d] | 0.5[c] | Not listed | EPA Group A[e] |
| Lithium arsenite | 72845-34-2 | 0.0003[f] | Not established | 0.01[f] | Not listed | EPA Group A[f] |

[a] EPA groups for carcinogenicity are classified as follows: EPA Group A: Human carcinogen; EPA Group B1: Probable Human Carcinogen - limited evidence in human studies; EPA Group B2: Probable Human Carcinogen - sufficient evidence from animal studies, inadequate evidence or no data from human studies; EPA Group C: Possible Human Carcinogen; EPA Group D: Not Classifiable as to Human Carcinogenicity.

[b] As cadmium.

[c] As chromium (VI).

[d] As chromium (VI) particulates.

[e] As chromium (VI) via the inhalation route of exposure. Carcinogenicity of chromium (VI) by the oral route of exposure cannot be determined and is classified as EPA Group D.

[f] As arsenic inorganic compounds.

Source: 29 CFR 1910; 40 CFR 355; NIOSH 1997; EPA 1999; ACGIH 1997.

Again, because of the nature and quantity of the chemicals at the HFBR, the effects of hazardous chemical exposure on workers is expected to be minimal.

## C.4 HUMAN HEALTH EFFECTS STUDIES: EPIDEMIOLOGY

Two recent epidemiological studies of the counties surrounding the BNL site have been conducted due to concerns regarding potential adverse health effects associated with the activities conducted at BNL. Most epidemiological studies of the populations living near a DOE site have been descriptive in nature and are what epidemiologists refer to as "ecologic" or "correlational" studies. Occupational epidemiological studies (that is, studies of workers) have been mostly analytic. A brief overview of epidemiology is presented in Section C.4.1. The two completed epidemiological studies related to the BNL site, along with their assumptions and limitations, are described in Section C.4.2, and several ongoing studies are summarized.

## C.4.1 BACKGROUND

Adverse health effects associated with ionizing radiation exposure were first identified about 60 years ago. Studies published in the 1930s first documented cancer among painters who used radium to paint watch dials from 1910 to 1920. Radiation therapy for disease has been used since the 1930s, and studies have shown that the risk of cancer is related to the amounts of radiation received. Nuclear weapons research and manufacture, and consequent exposure to radiation, began in the late 1930s. Exposure to radionuclides has changed over time, with higher levels occurring in the early days of research and production. Due to concern regarding potential adverse health effects, numerous epidemiological studies have been conducted among workers who manufactured and tested nuclear weapons. More recently, concerns about offsite radiological contaminants have resulted in health studies among communities that surround DOE facilities. The following section gives an overview of epidemiology followed by a review of epidemiological studies for the BNL site.

### C.4.1.1 Ecologic Studies

Ecologic studies compare the frequency of a disease in groups of people in conjunction with simple descriptive studies of geographic information in an attempt to determine how health events among populations vary with levels of exposure. These groups may be identified as the residents of a neighborhood, a city, or a county where demographic information and disease or mortality data are available. Exposure to specific agents may be defined in terms of residential location or proximity to a particular area, such as distance from a waste disposal site. An example of an ecologic study would be an examination of the rate of heart disease among community residents in relation to the quality of their drinking water.

The major disadvantage of ecologic studies is that the measure of exposure is based on the average level of exposure in the community, when what is needed is each individual's exposure. Ecologic studies do not take into account other factors — such as age, race, and the geographic history of the person — that may also be related to disease. These types of studies may lead to incorrect conclusions, known as "ecologic fallacies." For the above example, it would be incorrect to assume that the level of water hardness influences the risk of getting heart disease. Despite the obvious problems with ecologic studies, they can be a useful first step in identifying possible associations between risk of disease and environmental exposures. However because of their potential for bias, ecologic studies should never be considered as more than an initial step in an investigation of the cause of a disease.

### C.4.1.2 Cohort Studies

The cohort study design is a type of epidemiological study frequently used to examine occupational exposures within a defined workforce. A cohort study requires a defined population that can be classified as being exposed or not exposed to an agent of interest, such as radiation or chemicals that influence the probability of occurrence of a given disease. Characterization of the exposure may be qualitative (for example, high, low, or no exposure) or very quantitative (for example, radiation measured in rem, or chemicals in parts per million). Surrogates for exposure, such as job titles, are frequently used in the absence of quantitative exposure data.

Individuals included in the study population are tracked for a period of time (usually over years), and fatalities in the study population are recorded. Fatality rates for the exposed worker population are compared with fatality rates for workers who did not have the exposure (internal comparison), or are compared with expected fatality rates based on the U.S. population or State fatality rates (external comparison). If the fatality rates differ, an association is said to exist between the disease and exposure. In cohorts where the exposure has not been characterized, excess mortality can be identified. However, these fatalities cannot be automatically attributed to a specific exposure, and additional studies may be warranted. More recent studies have looked at other disease endpoints, such as overall and cause-specific cancer incidence (newly diagnosed) rates.

Most cohort studies at DOE sites have been historical cohort studies, that is, the exposure occurred some time in the past. These studies rely on past records to document exposure. This type of study can be difficult if exposure records are incomplete. Cohort studies require extremely large populations that have been followed for 20 to 30 years. They are generally difficult to conduct and are very expensive. These studies are not well suited to studying diseases that are rare. Cohort studies do, however, provide a direct estimate of the risk of fatality from a specific disease and allow an investigator to look at many disease endpoints.

#### C.4.1.3 Case-Control Studies

The case-control study design starts with the identification of persons with the disease of interest (case) and a suitable comparison (control) population of persons without the disease. Controls must be persons who are at risk for the disease and are representative of the population that generated the cases. The selection of an appropriate control group is often quite difficult. Cases and controls are then compared with respect to the proportion of individuals exposed to the agent of interest. Case-control studies require fewer persons than cohort studies, and therefore are usually less costly and less time consuming, but are limited to the study of one disease (or cause of fatality). This type of study is well suited for the study of rare diseases and is generally used to examine the relationship between a specific disease and exposure.

## C.4.2 EPIDEMIOLOGICAL STUDIES

This section discusses two recent epidemiological studies of the communities surrounding BNL, and several on-going studies are summarized. One recent study focused on breast cancer incidence rates (Sternglass 1994), while the other study investigated the incidence rates of a number of different types of cancers and congenital malformations (Grimson 1998). For the remainder of this section, these studies will be referred to as the “Sternglass study” and the “Grimson study.” Both of these studies were ecological studies, and thus the limitations discussed in Section C.4.1.1.1 apply.

The Sternglass study analyzed the pattern of breast cancer mortality rates in the New York metropolitan area including the Nassau and Suffolk counties on Long Island, which surround BNL. This study reviewed data from 1960 to 1987. The study found that the breast cancer incidence rate for all the community groups within 24 km (15 mi) of BNL was about 11 percent higher than Suffolk county as a whole.

The Grimson study was performed in response to allegations that BNL operations were causing harm to the environment or to the public health in communities in close proximity to BNL. This study included consideration of the results of the Sternglass study, and is the most recent information available. For this EIS, the Grimson study is considered the best available epidemiology report for the area surrounding BNL. The Grimson study was performed from October 1996 through January 1998 and considered data from the most up-to-date six year period, 1988-1993. The Grimson study investigated the area within a 24 km (15 mi) radius of BNL. Specifically, the study analyzed the geographic patterns of the following types of cancers: thyroid, leukemia, non-Hodgkin's lymphoma, brain and nervous system, female breast, prostate, liver, bone, multiple myeloma, and childhood rhabdomyosarcoma. The study also analyzed the geographic patterns of the following types of congenital malfunctions: major malformations (as a group), chromosome anomalies, Down's syndrome, tracheoesophageal fistula, hip dislocation, neural tube defect, anencephaly, spina bifida, microcephalus, and a group of rare mutations not inherited.

The Grimson study concluded that the cancer incidence rates of all types of cancers studied were not elevated near BNL, and the cancer incidence rates in the different quadrants surrounding BNL were not significantly different nor correlated with contaminated groundwater plume nor wind directions. The study also specifically concluded that the incidence of childhood rhabdomyosarcoma is not elevated in the area surrounding BNL. The study did find that the incidence rate of female breast cancer on the east end of Long Island (greater than 24 km [15 mi] east of BNL) was significantly higher than the rates of the areas adjacent to BNL. The study noted that the reason for this increase has not been specifically identified.

There are several other epidemiological studies currently in progress. The U.S. DOE Office of Epidemiologic Studies is working with the New York State Cancer Registry to measure the rate of all types of cancers among former and current BNL workers and compare these BNL worker rates with rates for New York State and Nassau and Suffolk counties. The Cancer Registry draft report is expected to be submitted in 1999.
In addition, the Agency for Toxic Substances Disease Registry (ATSDR) is evaluating the results of groundwater monitoring of onsite monitoring wells and offsite residential wells to determine whether residents are being exposed to contaminants at levels that could result in adverse health effects. This groundwater consultation is expected to be completed in 1999.

# C.5 FACILITY ACCIDENTS

## C.5.1 EVALUATION METHODOLOGIES AND ASSUMPTIONS

In order to support the purposes of NEPA review, this Appendix chooses a set of accidents and presents their potential frequencies and consequences based on existing information and on some reanalysis. The purpose of this section is to explain how existing information has been used, how the selection of accidents has been performed, and how and why certain reanalysis has been done.

The following circumstances either precluded or distorted comparisons of accidents across alternatives when accident analyses were based solely on information available when this EIS was initiated:

- Existing information sources did not reflect the current plant configuration. Systems have been added since some of the analysis was performed.
- Existing information sources did not reflect the best current understanding of how

certain accident sequences would evolve. New analysis has been warranted in these areas.

- The different information sources analyzed accident consequences using different criteria and methodologies, and no existing source presented one key dose measure used in this EIS ("mean" dose to maximally exposed individual).
- The frequencies of some accidents were overstated in some analyses as a result of intentional or unintentional conservatisms in the modeling.

Because of these circumstances, selection and reanalysis of accidents has been necessary in order to develop appropriate comparisons.

This appendix discusses the process of selection and reanalysis of accidents. This section furnishes background on the application of existing analyses to the EIS assessment. It begins by citing particular facets of DOE guidance to EIS preparers in the area of accident analysis. With this guidance as background, the discussion next turns to the pros and cons of the best-available previously existing sources of information on possible accidents at the HFBR. In a subsequent section, an overview of results from the existing *Probabilistic Risk Analysis* (PRA) is provided; then particular accidents are selected and discussed in more detail. Finally, a selection of accidents is made for purposes of performing a comparison of alternatives, and these results are then presented in Section 4.11.

**<u>NEPA Guidance On Selection of "Design Basis" and "Beyond Design Basis" Accidents</u>**

The following guidance is provided to EIS preparers (DOE 1993b):

> This [accident analysis] section deals with "environmental impacts that will not necessarily occur under a proposed action, but which are reasonably foreseeable. The term "reasonably foreseeable" has no precise definition. Its interpretation should be guided by two primary purposes of NEPA review: (1) to determine whether a proposed action has the potential for significant impacts (EA), and (2) to inform an agency (and the public) in making reasonable choices among alternatives (EA and EIS).
>
> For both purposes above, "reasonably foreseeable" includes impacts that may have very large or catastrophic consequences even if their probability of occurrence is low, provided that the impact analysis is supported by credible scientific evidence, is not based on pure conjecture, and is within the rule of reason. [...]
>
> For a proposed action that involves a facility or component with a set of design basis criteria (DOE 6430.1A), consider the following two major categories of accidents.
>
> > Within design basis: First focus on accident, failure, or error scenarios within the design basis and determine the type of event that is likely to cause the greatest consequences, supporting that determination with rough estimates of or qualitative judgments about the magnitude of the consequences. Typically, these events will have probabilities of greater than $10^{-6}$ per year, especially for natural phenomenon events.
> >
> > Beyond design basis: Look beyond design basis to see if there may be events of such large consequences that they need to be considered in order to satisfy the primary purposes of NEPA review as stated in the first paragraph in this section ['two primary purposes' quoted above]. Generally, examine the probability range $10^{-6}$ to $10^{-7}$ per year to the degree that events within this range bear on satisfying the two primary

> purposes of NEPA review cited above. As a practical matter (including litigation history), events with probabilities less than $10^{-7}$ per year will rarely need to be examined.

This guidance requires that events beyond the design basis be discussed in the EIS. It does not require presentation of an integrated risk profile, or the assessment of the total risk associated with all accidents in a particular severity category, or all possible accidents initiated by a particular event such as "loss of offsite power" (LOOP). Rather, it requires discussion of particular accidents, each one having a frequency greater than $10^{-7}$ per year, each selected for inclusion based on its consequences. The frequency criterion (greater than $10^{-7}$, or one in ten million per year) is applied not at the initiating event level (for example, LOOP) but at the full accident sequence level, including functional failures as well as the initiating event (for example, LOOP, followed by failure of all cooling, failure to recover power, and radiological release of a particular kind).

This appendix discusses the application of available information to the selection and characterization of the set of accidents discussed in Section 4.11. The information sources considered include the *Safety Analysis Report* (SAR) (some design basis events), various sections of the PRA performed for the HFBR (some design basis events and some beyond design basis events), information on modifications to the physical plant that have been carried out since the PRA was performed, and analysis performed since the PRA (some of it for purposes of this EIS). Because of the character of the information sources considered, and the obsolescence of some information, it has been necessary to choose carefully and to reanalyze some events, in order to present a collection of results in Section 4.11 that satisfy the purpose of NEPA review as articulated above. This appendix fully discusses the process.

**Existing Information on "Design Basis" Accidents: Safety Analysis**

In order to clarify what is meant by the phrases "design basis" and "beyond design basis," and explain what this implies about the results presented in the SAR, this subsection summarizes definitions appearing in DOE Standards and Orders, as well as the definition appearing in the Code of Federal Regulations (CFR). The SAR is then briefly discussed in light of these definitions.

Following are some illustrative definitions of "design basis" and related terminology:

> **Design basis accidents (DBAs)** means accidents that are postulated for the purpose of establishing functional requirements for safety significant structures, systems, components, and equipment. [DOE Order 5480.23]

> **Design basis accidents (DBAs).** Postulated accidents, or natural forces, and resulting conditions for which the confinement structure, systems, components and equipment must meet their functional goals. These safety class items are those necessary to assure the capability: to safely shut down operations, maintain the plant in a safe shutdown condition, and maintain integrity of the final confinement barrier of radioactive or other hazardous materials; to prevent or mitigate the consequences of accidents; or to monitor releases that could result in potential offsite exposures. [DOE Order 6430.1A]

> Design bases means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted

> "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. [10 CFR 50, Definitions]

Essentially, to say that an accident is "within the design basis" is to say that it has been allowed for in the design of the facility or that the design is capable of dealing with the accident. Consequences will be kept below levels specified in the applicable acceptance criteria. The set of design basis accidents is chosen in such a way that collectively, they pose stringent and diverse challenges to facility safety. Therefore, a system's ability to cope with them is strong evidence of good design. The SAR presents the analysis that shows that, as a result of design features, the consequences of the design basis accidents are acceptable, based on design attributes and operational practices that need to be followed in order to make the safety analysis accurate.

The safety analysis that establishes that the system can cope with design basis accidents is deliberately "conservative." An analysis is said to be "conservative" if its modeling assumptions (parameter values or other choices) tend to overstate the potential frequency or potential consequences of an accident. If a high level of assurance is desired demonstrating that the system performance is truly acceptable, then conservative analysis methods and a conservative acceptance criterion are employed so that the possibility of small deviations in the actual situation would not raise concerns about the finding of system adequacy.

The SAR is a key element in regulatory oversight of facility safety. The SAR shows the regulator that the design accommodates certain "design basis" events, so that even if those accidents occur, the facility is capable of withstanding them, in the sense that their consequences are low enough to satisfy applicable criteria. Because the oversight function relies on the analysis in the SAR, it is intended to be a robust demonstration of accident consequences; the analysis methods are deliberately conservative. This means that for other purposes, the SAR is not necessarily an appropriate statement of actual accident consequences. Even so, the analyzed consequences of events within the design basis are generally not very severe, because if they were severe the design would be modified to better cope with such events.

As previously noted, NEPA guidance requires EIS preparers to consider not only design basis events, but also "beyond design basis" events, which have not necessarily been allowed for in the design. These are discussed below.

**Existing Information on "Beyond Design Basis" Accidents: Probabilistic Risk Analysis**

DOE-STD-3009 defines "Beyond Design Basis Accident" as "[a]n accident of the same type as a design basis accident (e.g., fire, earthquake, spill, explosion, etc.), but defined by parameters that exceed in severity the parameters defined for the design basis accident." One example is a pipe break larger than the size considered in the design basis. Another kind of beyond-design-basis accident is a scenario involving an unlikely conjunction of multiple failures or an unlikely combination of events, such as a LOOP occurring during a hurricane just as a 7.0 earthquake hits.

NEPA guidance mandates consideration of more severe events that are beyond the design basis,

> … to see if there may be events of such large consequences that they need to be considered in order to satisfy the primary purposes of NEPA review …. Generally, examine the probability range $10^{-6}$ to $10^{-7}$ per year to the degree that events within this range bear on satisfying the two primary

> purposes of NEPA review…. As a practical matter (including litigation history), events with probabilities less than $10^{-7}$ per year will rarely need to be examined (DOE 1993b).

Clearly, the consequences of beyond design basis events may be significantly worse than the consequences of design-basis events, because beyond design basis events have not necessarily been allowed for in the design. They are not, however, automatically worse, partly because most nuclear facility designs incorporate significant margin and partly because the phenomenology (physics, chemistry, etc.) of the actual accident progression may simply not create significant consequences beyond a certain distance.

Information about beyond design basis accidents is developed in a PRA. A PRA produces a different kind of information about facility safety than is presented in a SAR. A PRA searches for all risk-significant scenarios to create an overall risk perspective, rather than focusing on a particular set of scenarios in order to establish a design's capability. A PRA considers not only events more severe than those postulated in establishing the design basis, but also events where the systems provided to cope with the design basis event fail to provide their design function.

The term "risk" has more than one definition. The risk associated with a given facility is sometimes defined as the sum, over all accident sequences, of the product of accident frequency (probability) and accident consequences (such as offsite dose). This formulation equates "risk" to "time-average consequences of operation." A more complex, but also widely-used, definition (Kaplan 1981) presents "risk" as a set of triplets:

- scenarios (what can go wrong, such as accident sequences leading to core damage)
- frequencies of those scenarios (usually quoted as 'events per year,' frequently interpreted as the chance of that scenario occurring in a given year)
- consequences of those scenarios (such as dose to offsite public if the scenario actually occurs)

This latter definition presents "risk" as a multi-dimensional set of facts, rather than a single number for each consequence type considered (dose, number of fatalities). In a sense, this definition speaks more directly to the NEPA review process, which does not require a complete risk analysis, but does call for a scenario-based comparison of alternatives rather than simply presenting an average annual offsite dose. However, many analysts use the time-averaged-consequences definition of risk to determine how far a risk analysis needs to go in pursuit of less-likely but possibly more-severe accident sequences. Basically, a PRA is not considered complete until it has identified and analyzed all accident sequences that can make a relatively significant difference to average consequences. For most PRA purposes, accident sequences need not be considered if their likelihood or consequences are so low that they cannot add much to the total average consequences.

It is found for many nuclear facilities that beyond-design-basis scenarios are important contributors to time-averaged consequences for some consequence types. As explained above, the consequences of within design basis scenarios are relatively low; this is the point of having those events in the design basis. The consequences of beyond design basis events can be worse, because the design did not necessarily allow for them; worse consequences may offset lower probabilities sufficiently to make beyond design basis events more significant contributors to risk. The sense of the NEPA guidance quoted above is that the purposes of NEPA review require some perspective on the consequences of such beyond design basis accidents, but not for extremely unlikely accidents (those whose potential frequencies are smaller than $10^{-7}$ per year).

**Definition of "Core Damage"**

Although radionuclides are present in relatively small quantities outside of the core (such as tritium in the coolant of the HFBR), most of the radionuclide inventory at a reactor is in the irradiated fuel elements. Most accidents having appreciable dose consequences to the general public involve "core damage" — overheating (melting) of these irradiated fuel elements — and subsequent release of these radionuclides to the atmosphere.

Not all core damage accidents have significant consequences to the public; it is possible to damage the core without releasing significant quantities of radionuclides to the atmosphere. Nevertheless, many reactor PRAs begin by analyzing core damage accidents, presenting their frequencies and causes, and then continue by analyzing the subsequent evolution of core damage accidents. One reason for this is that if significant core damage occurs, even in the course of an incident having no offsite consequences, there is likely to be a very significant interruption in the service provided by the facility; the facility operator therefore has reason to be interested in core damage for its own sake. Another reason is that the severity of possible radiological consequences is relatively limited unless core damage occurs. Finally, presentation of results both at the core damage level and at the release level provides useful insights into why releases occur for some accidents and not others, the mechanisms of release, how release might be mitigated for particular core damage accidents, and so on.

In addition to the fuel handling accident, the HFBR PRA (BNL 1993b) considers two levels of core damage.

> Core damage can be divided into two parts: "early" core damage, where the "hot spots," about 1% of the core, should experience some melting soon after the initiator; and major core damage, where the total core is assumed to experience some level of melting. Because of the specifics of the system, it is assumed that following an initiator, the plant can end up in only one of three states: 1) the accident is controlled and no release occurs; 2) the core experiences early damage and the accident is mitigated with some release; and 3) the entire core experiences melting. Early core damage can occur when there is a loss of forced flow cooling for a set amount of time (3 minutes after scram for 60 MW operation and 50 seconds after scram for 40 MW) since the decay power is high enough that the natural circulation cooling, established when the flow reversal valves have opened, is insufficient to adequately cool the hot spots in the core.

Note that at the time the HFBR PRA was performed, the assumption was that 3 minutes of forced flow were needed after scram at 60 MW. As discussed in subsequent subsections, work performed since the HFBR PRA was written has led to a significant modification of that requirement (see sections C.5.1.1.2.2, C.5.1.1.2.3, and C.5.1.1.2.4). However, for extremely large breaks, it appears that "early" core damage can still occur at 60 MW (Palmrose 1999). As a result of the accident selection process described in the following subsections, accidents were selected representing all of the core damage categories defined above: no damage, early damage, and entire core damage.

**Treatment of Uncertainties**

In the phrase "probabilistic risk assessment" (PRA), the term "probabilistic" refers to the PRA treatment of uncertainties. There can be very large uncertainties surrounding many aspects of a PRA, and decisions based on PRA results need to take account of this circumstance.

For example, there may be several orders of magnitude (factors of ten) uncertainty in the assessment of the frequency of a particular failure event, such as a large rupture of a specific pipe. Because it is the point of PRA to achieve a comprehensive statement of risk, PRA analysts are not free to avoid discussing an event just for lack of information. If an event has the potential to contribute significantly to average consequences, the analysts need to characterize the frequency and consequences of that event to the extent possible within their state of knowledge. Therefore, in order to obtain a result, some PRAs (1) define a range of possible values for each highly uncertain variable, (2) impute state-of-knowledge probabilities to each possible value (for instance, they assess how likely it is that each possibility is the right one), and (3) average the PRA results over this set of probability-weighted parameter values. The outcome of such an approach is not simply a statement about the system; it is also a statement about the analysts' state of knowledge of that system. A PRA therefore not only provides estimates of "risk" but also indicates where more work should be done. This is discussed later, because it has affected the choice of PRA events selected for comparison of alternatives.

Large uncertainties typically do not arise in SAR analysis because large uncertainties arising within the design basis obviously need to be eliminated by design, experiment, or further analysis and the SAR analysis is performed to conservatively bound any remaining uncertainties. Uncertainties arise in PRAs because the events being addressed are outside the normal operational envelope and, in some cases, beyond engineering experience.

## Conservatism in PRA

While a SAR is supposed to be conservative in order for its findings to be robust, the situation is different for a PRA. For most purposes, risk analysis is supposed to be unbiased (neither conservative nor optimistic). Frequently, a PRA is performed to establish whether modifications to a facility (improvements to a design, or "backfits" to an existing facility) would be cost-beneficial. This can be done by comparing the cost of the modification to the monetary value of the change in average consequences, computed before and after the modification. In such an application, conservatism in a PRA is a mixed blessing: systematic overstatement of risks could create false issues and tend to lead to wasting resources, while uneven conservatism could divert resources from more significant issues to less significant ones, or distort comparison of EIS alternatives. On the other hand, if the apparent finding is that all risks are so low that no backfits are cost-beneficial, then demonstrable conservatism in the analysis adds to the robustness of the finding. Indeed, at the time of the HFBR PRA, conservatism in the PRA was not an issue because the major result of the PRA was the conclusion that the HFBR meets applicable DOE safety goals by a wide margin:

> The final results of the consequence analysis indicate that the risks associated with the operation of the HFBR are more than two orders of magnitude below the safety goals set by the DOE for the operation of its facilities. The DOE has set these goals as being 0.1 percent of the probability per year to the general public that a fatality will occur from an accidental death or death due to cancer because of the operation of the facility. Numerically, the safety goal for individual early fatalities (one to two miles) is $5x10^{-7}$/ry ["reactor year," or year of reactor operation] and the safety goal for the individual latent cancers (one to ten miles) is $2x10^{-6}$/ry. For operation of the HFBR, the population weighted early fatalities were calculated to be less than $1x10^{-11}$/ry for both 40 MW and 60 MW operation. The population weighted latent cancers were

calculated to have a mean value of $8x10^{-10}$/ry and $1x10^{-9}$/ry for 40 and 60 MW operation, respectively. Because the HFBR is presently being operated at 30 MW, an extrapolation (since a detailed Level 1 PRA was not performed for this power level) to that power level was performed. The population weighted latent cancers were then estimated to have a mean value of $7x10^{-10}$/ry.

In addition to indicating that safety goals were satisfied, the PRA results indicated that relatively little in the way of backfits would be cost-beneficial because there is very little risk present to be averted. In short, the overall risk was considered to be so low that there was no incentive at the time of the PRA to reconsider any conservatisms present in the analysis of any particular event. As indicated previously, however, uneven conservatism cuts unevenly across EIS alternatives and severely complicates a comparison of alternatives (unless the risks of all alternatives are deemed so low as to be negligible, and therefore effectively equal for purposes of comparison).

**Summary**

In light of the above, and taking into account the NEPA guidance and the available sources of information, the comparison has been developed in the following way. Details are presented in this section and in background reports.

- A full spectrum of events has been considered, ranging in potential consequence and potential frequency from minor incidents that have already occurred, through design basis events addressed in the SAR, and on out to full-scale core damage events. These events were identified based on best-available existing work (SAR, PRAs, operating experience).

- Within this spectrum of events, based on information available, important categories of events have been identified for use in comparison of alternatives. The categories are based on consequences. The categories are: major core damage with a breached confinement, major core damage with an essentially intact confinement, minor core damage with an essentially intact confinement, and events involving no significant core damage.

- Events have been selected in each category whose consequences and frequency can both be characterized well enough for present purposes, based either on existing information or on reanalysis considered practical at present. Here, “event” means specification of both an accident initiator (such as pipe rupture) and a particular combination of functional successes and failures, leading to a reasonably well-defined physical outcome (such as “major core damage”). For example, the LOOP *accident* analyzed here is not just a LOOP (which has been known to occur), but rather a LOOP with protracted loss of the cooling function, failure to supply makeup, and failure to recover offsite power for a long time.

*Characterization of Accident Consequences:* Neither the SAR nor the PRA furnished estimates of mean dose to maximally exposed individuals (MEI), so for selected accidents, reanalysis of dose consequences was performed in order to characterize dose to the MEI in a consistent way across accidents and across alternatives. Apart from addressing MEI doses, evaluating 30 MW operation, and using up-to-date core inventory information, this analysis was generally consistent with the HFBR PRA.

*Characterization of Accident Frequencies:* Evaluation and reanalysis of the PRA scenarios has been required in order to choose accidents whose potential frequencies can be characterized appropriately for purposes of comparison of alternatives.

NEPA guidance does not call for presentation of a complete risk analysis, or even an estimate of total accident frequency on an absolute basis. Significant work has been necessary in order to characterize selected accidents for purposes of this EIS, and a great deal more would be needed to characterize the total frequency. The choice of events chosen for presentation does, however, fairly illustrate the range of consequences analyzed in the PRA.

Some effort has been required to derive needed information from the PRA reports. One reason for this is that a major purpose of the PRA analyses was to assess the HFBR with respect to DOE safety goals for facilities of this type. For that purpose, frequency-weighted consequences are derived and presented in the PRA, but certain information about individual accident sequences (such as dose to the MEI) was not needed and was not presented in the PRA. The PRA also did not analyze 30 MW operation at all (although it analyzed 40 MW operation), and the PRA did not analyze consequences of certain events of interest (such as the fuel handling accident which is discussed in the SAR).

The following subsections discuss the re-analyses performed for this EIS. First, in order to provide some risk perspective and show the starting point of the present evaluations, the overall PRA results are summarized (C.5.1.1.1). This is followed by a discussion of conservatisms and uncertainties in the analysis of several accident scenarios, some of which were reassessed in order to support a comparison of the alternatives (C.5.1.1.2). Finally, in order to put comparisons of event consequences across alternatives on a consistent basis, consequence analyses were performed for several events. These calculations are discussed in C.5.1.1.3.

## C.5.1.1 Introduction

### C.5.1.1.1 *Overview of PRA Results*

This section provides an overview of the major results of the PRA that was performed on the HFBR over several years (BNL 1990a, BNL 1990b, BNL 1993b, and BNL 1994). This analysis addressed core damage scenarios initiated by both internal and external events, and analyzed scenarios including public health consequences. The Level 1 PRA analysis [BNL 1990a, BNL 1990b], which analyzes scenarios from the point of initiation through the point of core damage, used the event tree and fault tree methodologies that are customarily applied in PRAs of reactors. Data gained in the course of operating the HFBR were factored into the analysis to the extent possible. The Level 2 PRA analysis (BNL 1993b, BNL 1994), covering scenarios from the point of core damage out to the point of release from confinement, made use of available experimental information to derive estimates of radionuclide releases from the core. For each scenario, the Level 2 PRA analysis explicitly addressed the efficacy of the confinement filtration function in reducing environmental releases based on the conditions that prevail in each scenario. The Level 3 PRA analysis (BNL 1993b, BNL 1994), which analyzed radiological health consequences, was based on a now-standard PRA computer code called “MELCOR Accident Consequence Code System” or MACCS (SNL 1990a, SNL 1990b, and SNL 1990c). It made use of available information on population density surrounding the HFBR, as well as available information on site meteorology, which would strongly affect the transport of radionuclides released in the course of an accident. Care was taken to ensure that the methodologies used in the PRA conformed to the BEIR V recommendations regarding modeling of health consequences. (BEIR V methodologies were discussed in Section C.2.1.2 of this appendix.)

The results provided in the PRA were the starting point for the process of selection and characterization of the accidents in this EIS. The PRA was performed essentially to compare the HFBR risk with DOE safety goals. This objective strongly affected the PRA's analytical approach. It has been previously mentioned that the HFBR risk is far below safety goal target levels; this means that for purposes of the safety goal comparison, the analysts could afford to make relatively conservative assumptions without affecting the conclusion that the HFBR meets safety goals. To put it another way, in the interest of saving time, the analysts chose to introduce simplifications that overstate risk in some scenarios, because the risk was so much lower than safety goal targets that the overstatement did not matter. However, overestimating scenario frequencies severely complicates the process of comparing EIS alternatives, especially the comparison of operating alternatives with non-operating alternatives.

The Internal Events portion of the PRA produced the assessment shown in Tables C.5-1, C.5-2, and C.5-3.

These tables show that the largest contributors to core damage frequency (CDF) from internally initiated accidents are not the largest contributors to offsite consequences. There are many reasons for this. One of the most important reasons is that in the analysis of the physics of accident progression, it was determined that many of the scenarios originally classified as "core damage" scenarios are not, in fact, core damage scenarios unless additional failures occur. This is explained in the following excerpt and discussed extensively in Section C.5.1.1.2.1:

> The LOOP initiators increase in the relative contribution to the consequences and the ATWS [anticipated transient without scram] and LLOCA [large loss-of-coolant accident] initiators decrease. The reason for this is that the SLOCA [small loss-of-coolant accident] and LLOCA have low probabilities of actually ending in core melt. The refinement of the ATWS sequences resulted in the conclusion that not all ATWS scenarios will lead to complete core melt. However both the LOOP and BTR [beam-tube rupture] will always lead to core melt [that is, the scenarios previously analyzed as leading to core damage are still analyzed as leading to core damage, because in these scenarios, the experimental facilities cooler cannot help]. One of the main reasons for the SLOCA and LLOCA being reduced in the consequences is the heat removal capability of the experimental facilities cooler. (BNL 1993b)

***Table C.5-1. "Major" Core Damage Frequency by Initiator Category, Internal Events***

| **Initiator Category** | **60 MW (events per yr)** | **40 MW (events per yr)** | **< 40 MW (events per yr)** |
|---|---|---|---|
| LOOP | $5.29x10^{-5}$ | $1.61x10^{-5}$ | $1.61x10^{-5}$ |
| Small LOCA[a] and cooler tube rupture | $7.11x10^{-5}$ | $4.81x10^{-5}$ | $4.81x10^{-6}$ |
| Medium LOCA | $1.45x10^{-7}$ | $9.88x10^{-8}$ | $9.88x10^{-8}$ |
| Large LOCA | $2.06x10^{-4}$ | $1.35x10^{-4}$ | $6.37x10^{-6}$ |
| BTR | $1.01x10^{-4}$ | $1.01x10^{-4}$ | $4.30x10^{-6}$ |
| Total major core damage | $5.05x10^{-4}$ | $3.69x10^{-4}$ | $9.99x10^{-5}$ |

[a] Loss of Coolant Accident

Note: No credit is reflected in these numbers for the ability of the experimental facilities cooler to maintain core cooling.

Source: BNL 1990a.

***Table C.5-2. "Minor" Core Damage Frequency by Initiator Category, Internal Events***

| **Initiator Category** | **60 MW (events per yr)** | **40 MW (events per yr)** | **< 40 MW (events per yr)** |
|---|---|---|---|
| Pump Seizure | $5.6x10^{-4}$ | NI | NI |
| Light water flooding of thimbles | $1.3x10^{-4}$ | $1.3x10^{-4}$ | $1.3x10^{-4}$ |
| Auxiliary rod break | $2.1x10^{-6}$ | NI | NI |
| Flow blockage | NA | NA | NA |
| Refueling discharge accident | $6.0x10^{-5}$ | $3.5x10^{-5}$ | $3.5x10^{-5}$ |
| Total minor core damage | $7.6x10^{-4}$ | $1.9x10^{-4}$ | $1.9x10^{-4}$ |

Note: NI = No Impact (that is., core damage does not result from postulated event at low power levels); NA = Information Not Available. The flow blockage results have been transcribed from Table 4-8 in BNL 1990a, although Table 6-1 of Volume 2 seems to indicate that the proper result is $2.9 \times 10^{-15}$. The frequency for the refueling discharge accident for power levels lower than 60 MW has been scaled from the 60 MW result based on the relative number of fuel elements handled at the lower power levels. At 30 MW, the frequency of the refueling discharge accident is $2.6 \times 10^{-5}$.

Source: BNL 1990a.

***Table C.5-3. Latent Cancer Fatalities by Initiator Category, Internal Events***

| | **60 MW Operation** | | | **40 MW Operation** | | |
|---|---|---|---|---|---|---|
| **Initiator Category** | **% of CDF[a]** | **% Onsite LCFs** | **% Offsite LCFs** | **% of CDF** | **% Onsite LCFs** | **% Offsite LCFs** |
| ATWS | 13 | 9 | 7 | 17 | 17 | 11 |
| LOOP | 10 | 25 | 24 | 4 | 14 | 11 |
| Small LOCA | 13 | 4 | 3 | 13 | 0 | 0 |
| Large LOCA | 42 | 17 | 17 | 37 | 6 | 2 |
| Beam Tube Rupture | 20 | 45 | 48 | 27 | 63 | 75 |
| Other Transient | 2 | NA[b] | NA | 2 | NA | NA |

[a] Table entries under "% of CDF" (core damage frequency) denote the percentage of the total computed core damage frequency attributable to the corresponding initiating event category; table entries under "LCF" denote the percentage of the computed frequency-weighted latent cancer fatalities attributable to the corresponding initiator category.

[b] NA = Information Not Available. The other transients were not analyzed in the PRA to determine onsite and offsite latent cancer fatalities because none of these transients represent a disproportionate level of risk with respect to the CDF. The "% Offsite LCFs" columns do not add up to 100 because of rounding errors and because the contribution from other transients has not been determined.
Source: BNL 1993b.

The situation is even more pronounced for external event initiators, as seen in Tables C.5-4 and C.5-5.

***Table C.5-4. Contributors to CDF and Latent Cancer Fatalities (60 MW Operation)***

| Initiator | CDF (1/yr) | Onsite LCFs (1/yr) | Offsite LCFs (1/yr) |
|---|---|---|---|
| Seismic Event | $3.8x10^{-4}$ | $7.5x10^{-5}$ | $1.8x10^{-3}$ |
| Fire / Flood | $4.1x10^{-5}$ | $3.9x10^{-6}$ | $1.4x10^{-4}$ |
| Aircraft Crash[a] | $2.1x10^{-6}$ | $3.0x10^{-5}$ | $2.3x10^{-4}$ |
| Severe Wind and Tornado | $4.0x10^{-5}$ | $9.3x10^{-5}$ | $4.5x10^{-3}$ |

[a] The analysis of a potential aircraft accident is discussed in section C.2.3.3
Source: BNL 1994.

***Table C.5-5. Contributions to Latent Cancer Fatalities by Initiator Category, External Events***

| | 60 MW Operation | | |
|---|---|---|---|
| Initiator Category | % of External Events CDF | % Onsite LCFs | % Offsite LCFs |
| Seismic Event | 81.6 | 36.7 | 27.9 |
| Fire and Flood | 9 | 2 | 2.1 |
| Aircraft Crash | 0.6 | 15.2 | 3.5 |
| Severe Wind and Tornado | 8.8 | 46.1 | 66.5 |

Note: Table entries under "% of External Events CDF" (core damage frequency) denote the percentage of computed CDF from external events that is attributable to the corresponding initiating event category; table entries under "LCF" denote the percentage of the computed frequency-weighted LCFs attributable to the corresponding initiator category.
Source: BNL 1994.

The mean per-year offsite consequences, reflecting both the likelihood of accidents and the consequences of those accidents, are shown in Table C.5-6.

***Table C.5-6. Latent Cancer Fatalities by Power Level and Initiator Category***

| | 40 MW Internal Events | 60 MW Internal Events | 40 MW External Events | 60 MW External Events |
|---|---|---|---|---|
| LCFs Per Reactor-Year (0-80 km) | $6.0x10^{-4}$ | $1.4x10^{-3}$ | $4.5x10^{-3}$ | $6.6x10^{-3}$ |

Source: BNL 1993b; BNL 1994.

In the above tables, columns labeled "% of CDF" are provided to show what fraction of CDF is due to events of a particular type. In Table C.5-3, for example, the category "Other

Transient" is seen to contribute only two percent to total CDF. This kind of information shows how "representative" particular kinds of events are, and serves to motivate the selection of accidents to be characterized.

The purpose of the following discussion is to choose a subset from these accidents upon which to base a sensible comparison of alternatives. As seen in the following subsections, some of these results almost surely overstate the risks because improvements have been made to the HFBR and because new information (such as the understanding of how accidents occur and progress) has become available that would justify less conservative analysis. The approach taken is to address the accident types that seem to make the largest contributions to frequency-weighted consequences, looking for scenarios that are not only representative of the risk profile but also representative of the variation in potential frequency and potential consequence with power level, and whose interpretation is not strongly affected by uncertainty in the physical models.

To this end, Table C.5-7 provides representative high-consequence and medium-consequence scenarios analyzed in the PRA, and Table C.5-8 provides representative low-consequence scenarios analyzed in the PRA and SAR. The scenarios in these tables will be discussed below in order to assess their usefulness for purposes of comparing alternatives.

The consequences of these accidents are related to the magnitude of the radiological release inside the confinement structure and to the effectiveness of the filtration function that removes radioactive material from the confinement atmosphere before it is released to the environment. Before being released to the environment, the confinement atmosphere is passed through high efficiency particulate air (HEPA) and charcoal filters, which are designed to be highly effective in removing particulates and certain forms of iodine. However, "[t]here is great concern about the reliability of both HEPA and activated charcoal filters when exposed to not only fog, but also high humidity…" (BNL 1993b). Accidents involving the full core typically generate a significant amount of steam, and filtration of those releases would be degraded due to the effect of steam on HEPAs. This consideration is reflected in consequence estimates presented in the PRA. Full core melts into an intact confinement structure are therefore "partially filtered." The fuel handling accident (FHA) and experimental materials accidents would be filtered, because there would be no large amount of water vapor to degrade the filtration function. The severe wind accident causing a release into a breached confinement and possible release to the environment would therefore be unfiltered. The $D_2O$ release would be outside of the confinement structure, and therefore not involve the confinement filter system.

***Table C.5-7. Representative High-Consequence and Medium-Consequence Scenarios Analyzed in PRA***

| Scenario Initiator | General Description of Scenario Modeled in PRA | Variation with Operating Power Level | Changes In Physical Plant or Technical Basis Since PRA was Performed |
|---|---|---|---|
| Large LOCA (PRA) | Large break, premature loss of forced flow as a result of break size, potentially resulting in core damage. Accident may be arrested by experimental facilities cooler | Some variation in initial radionuclide inventory.<br><br>Break size above which forced flow is required varies with operating power level; | Further analysis on flow reversal dynamics establishes that some breaks previously considered "large" are now mitigable without forced flow, warranting some reduction in core damage |

| | | | |
|---|---|---|---|
| | (EFC).<br><br>Release of material outside of the confinement structure is partially filtered. | therefore, frequency of LOCAs falling in this category varies with operating power level. | frequency assessed at 60 MW.<br><br>All LOCA scenarios at all power levels can be stabilized by the EFC at or before minor core damage, provided that the EFC is normally operating at increased flow rate before the LOCA. |

***Table C.5-7. Representative High-Consequence and Medium-Consequence Scenarios Analyzed in PRA—Continued***

| Scenario Initiator | General Description of Scenario Modeled in PRA | Variation with Operating Power Level | Changes In Physical Plant or Technical Basis Since PRA was Performed |
|---|---|---|---|
| LOOP (PRA) | LOOP, normal cooling function not available, core water inventory not replenished; core damage occurs. Release of material outside of the confinement structure is partially filtered. | Some variation in initial radionuclide inventory.<br><br>Some variation in accident timing (end point reached more slowly at lower power even if operator takes no action). Affects event frequency and release.<br><br>Time limits on operator actions are more forgiving at lower power levels (more time available to take action) | Addition of seismically qualified poison water and seismically qualified light water makeup may reduce likelihood of core damage in the event the original systems fail. Human error probabilities were excessively conservative. Consideration of reactor operating cycle as well as additional ten years of data affects frequency of initiator. |
| BTR (PRA) | Very large postulated break occurs suddenly in beam tube; coolant is expelled, core damage occurs early relative to other severe scenarios. Release of material outside of the confinement structure is partially filtered. | No variation in assessed frequency of accident with power level.<br><br>Some variation in initial radionuclide inventory.<br><br>Some variation in accident timing (damage proceeds more quickly at higher power). | More analyses and testing done to verify satisfactory condition of beam tubes. See Section C.5.1.1.2.3. |
| SWT (PRA) | Event causes LOOP, breaches confinement with projectile and also eliminates then-existing coolant makeup. Release of material not filtered because confinement structure is breached. | No variation in assessed frequency of accident with power level.<br><br>Some variation in source term. | Addition of seismically qualified poison water and seismically qualified light water makeup may reduce likelihood of core damage given the level of damage postulated in the PRA's event description. Human error probabilities assigned in the PRA were excessively conservative. The PRA's estimated projectile damage probabilities have been updated to account for double counting of damage to adjacent components, |

| | | | |
|---|---|---|---|
| | | | elimination of projectile damage to system that fails but does not increase the likelihood of an accident or its consequences, and error in penetration probabilities. The PRA’s assessment of tornado hit frequency did not consider reactor operating cycle and is inconsistent with basis for projectile probabilities and observed experience. See Section C.5.1.1.2.4. |

***Table C.5-8. Representative Low-Consequence Scenarios***

| Scenario Initiator | General Description of Scenario Modeled in PRA or SAR | Variation with Operating Power Level | Changes In Physical Plant or Technical Basis Since PRA was Performed |
|---|---|---|---|
| FHA (SAR, PRA) | Failure of successful transfer of fuel element from vessel to spent fuel pool, leading to overheating of that one element. Release of material outside of confinement structure is filtered. | The estimated consequences have minimal variation with operating power level. Refueling operation is delayed to minimize source term and is delayed longer if operation is at higher power. Frequency is lower at lower power levels because fewer fuel elements per year are handled. | None |
| Accident involving handling of experimental material (Not within scope of SAR or PRA; see note below.) | (Generic) Research programs conducted at the HFBR sometimes involve radioactive material, creating hypothetical potential for radiological incidents involving quantities of material much smaller than accidents involving reactor fuel. Ex-confinement release, if any, is filtered. | None | Proactive safety review of such operations enhanced after the fire at the TRISTAN experiment |
| $D_2O$ Release (SAR) | Primary coolant leaks through heat exchanger, reaches recharge basin without being detected. Confinement is not a factor. An insignificant airborne release would occur due to evaporation. | No significant variation unless the allowable tritium burden of coolant varied with power level, which is not the case. | None |

Note: The SAR addresses HFBR operations, but not all aspects of experimental programs. This kind of accident is not part of SAR accident analysis because it is bounded by other analyzed accidents. For reasons explained under “Definition of Core Damage,” in Section C.5.1.1, the PRA did not address this class of event either.

#### C.5.1.1.2 *Selection of Suitable Scenarios to Support Comparison of*

### *Alternatives*

Significant changes have taken place in the technical basis for evaluation of certain scenarios since the PRA was performed. One significant instance is the present understanding of flow reversal, discussed below in Section C.5.1.1.2.1. It is seen that the assumptions made in the PRA are very conservative. Not reflecting the current understanding would skew the comparison. Significant changes also have been made in the physical plant itself, changing some of the accident frequencies. Again, not reflecting the current situation would skew the comparison.

Additionally, some of the accident sequences in the original PRA are sufficiently uncertain and sufficiently debatable (such as the BTR) that they are less suitable for present purposes than accident sequences that are less uncertain. These sequences provide a less robust basis for comparison of alternatives.

In light of the above, two scenarios that seemed initially to address most of the above issues are the severe wind/tornado (SWT) scenario and a LOOP scenario. Specifically, based on consequences outside of the reactor confinement, the SWT event was assessed in the PRA as dominant on the basis of frequency-weighted consequences, while the LOOP event was a relatively significant contributor to frequency-weighted consequences and generally representative of severe accidents other than SWT (for example, BTR), for ex-facility consequences. Modifications to the physical plant and issues with the original analysis caused a re-evaluation of these events to be undertaken, with the result that their frequencies have been very substantially modified. These reanalyses are summarized below in Sections C.5.1.1.2.2 and C.5.1.1.2.4. As a result of the reanalysis, the SWT event now has a much lower frequency, but its consequences are still relatively significant. It is therefore an appropriate choice as the representative accident in the most severe category (complete core damage, breached confinement). The LOOP accident's frequency is reduced from the original estimate, but its consequences are still representative of the next most severe category (complete core damage, unbreached confinement). LOCAs, if allowed to proceed to full core damage, would have consequences similar to those presented for LOOP, but the present understanding of their phenomenology indicates that even the largest breaks lead only to minor core damage. Correspondingly, a break of this kind is selected to represent that category.

### *C.5.1.1.2.1 Large Loss-Of-Coolant Accident (LOCA)*

The large LOCA scenario has received attention for some years, partly because of certain special characteristics of the HFBR. Normal, forced-flow core cooling in the HFBR is downward through the core, but in some scenarios it is necessary to make a transition to natural circulation cooling, in which flow would be upward through the core. If the core is producing too much heat when it goes through the transition to natural circulation cooling from one flow direction to the other, core damage results. The conditions under which core damage might occur depend on how quickly the transition occurs after the initiating event (which depends on several things, including break size), how much heat the core is producing (which depends on the operating power level and how long the core has been at power), and many details of thermal-hydraulics.

When the PRA was performed, it was assumed that any break at 60 MW having an equivalent diameter greater than about 2 cm (0.8 in) would lead to core damage as a result of premature loss of forced flow (BNL 1990b). The frequency of this class of breaks was quantified based on an assessment of how much piping there is in the system in the relevant size and elevation category, together with a simple, widely used model providing estimates of break frequencies per unit piping length. The pipes contributing to the frequency assessment of large LOCA are listed in Table 4-6 of Vol. 2 of the PRA (BNL 1990b). On page 5-7 of BNL 1990a, the contribution of large LOCA at 40 MW was quantified from Table 4-6 cited above, by noting that at 40 MW, the break needs to be greater than about 7 cm (2.8 in) in diameter instead of about 2 cm (0.8 in) in diameter in order to cause core damage. Counting only those contributions from Table 4-6 that satisfy this criterion, the analysts obtain the "large LOCA" contribution at each power level, and present the comparison between 60 MW and 40 MW summarized above on the core damage frequency table.

At the time the PRA was performed, its assumption regarding the need for forced flow was consistent with the technical basis that existed at the time. Since then, significant work has been done to improve the technical understanding of core cooling in these scenarios (BNL 1997a). The current accident analysis analyzes a break in the primary system of 9 $cm^2$ (1.4 $in^2$), a size that is dictated by ANS/ANSI guidance for medium energy systems such as the HFBR (BNL 1998). Analysis performed shows that the consequences of this break are acceptable (that is, there is no core damage) at 60 MW operation. In addition, analysis also shows that at the same break size, the power level could be increased by over a factor of two without violating the core damage criterion used in the analysis. A comparison of the PRA assumption with the analysis is shown in Table C.5-9 (BNL 1998).

***Table C.5-9. Comparison of PRA Assumption (1990) With Accident Analysis (1998)***

| Definition of Large LOCA in PRA (Size above which core damage would occur as a result of loss of forced flow, 60 MW Operation) | LOCA Size Demonstrated Acceptable at 128 MW |
|---|---|
| 2.06 cm diameter (3.3 $cm^2$) | 3.38 cm diameter (9 $cm^2$) |

Not established by these analyses is the maximum break size consistent with no core damage at 60 MW. Also not established is the maximum operating power that would be consistent with no core damage without any forced flow at all (BNL 1990a).

Safety analyses for commercial light-water reactors normally analyze "large LOCA" as a complete break in the largest pipe, and show that plant systems are capable of achieving acceptable accident mitigation despite the occurrence of a LOCA of that size. The performance required for an event of this severity is mandated by code, which takes into account the significantly higher operating pressures and temperatures of commercial reactors. PRAs for commercial plants either stop at this break size (typical) or mention the possibility of larger breaks (either a vessel rupture or perhaps failure of some other passive component), assigning relatively very low frequencies to such possibilities.

In the HFBR, the situation is different. The analysis of a complete break of the largest pipe is not required for the SAR; instead, in recognition of the HFBR being a lower-energy system, the safety analysis evaluates a smaller break (9 $cm^2$ [1.4 $in^2$]) consistent with ANS/ANSI guidance. This break size is shown to be mitigable at all operating power levels under consideration. The PRA, having a different mandate, considered a larger break size, and quantified the frequency using such methods as were available and capable of application within the resource constraints that prevailed at the time. The PRA states (BNL 1990a):

> The frequency of large LOCAs used in HFBR-PRA was calculated based on the

> Thomas model. As discussed earlier in Section 3.5 of this volume, the transition probability of leak to break recommended by the Thomas model is quite conservative. Future work will be aimed at obtaining a better estimate of large LOCA frequency by using the Paris model which is a more deterministic model compared to the empirical Thomas model. …

Results obtained in the Level 2 (BNL 1993b) (physics of accident progression and release into confinement) and Level 3 (radionuclide transport) analyses further complicate the comparison between large-LOCA-induced CDFs at different power levels, because the Level 2 PRA essentially revised the Level 1 results, taking credit for EFC cooling (which is discussed below). Based on the understanding current in 1993, there is an 81 percent chance that a given large LOCA occurring during 60 MW operation would be stabilized after "early" core damage, and a 90 percent chance at 40 MW (BNL 1993b). One effect of this is that (as demonstrated in Table 7-4 of BNL 1993b) one sees that the majority of large LOCA frequency is associated with a radiological release whose offsite impact (in terms of number of LCFs) is roughly one percent of the impact of most core melts analyzed (at 40 MW, 0.05 LCFs for most large LOCA versus 4 to 6 LCFs for a typical core melt, and at 60 MW, 0.08 LCFs for most large LOCAs versus 6 to 8 for a typical core melt). This means that even without the new SAR results, the comparison between 60 MW and 40 MW is more complicated than suggested in the CDF comparison based on Table 4-8 of BNL 1990a and summarized above. As a further explanation of this reduction in core melt frequency, BNL 1993b states: "…the SLOCA [small LOCA] and LLOCA [large LOCA] have low probabilities of actually ending in core melt… One of the main reasons for the SLOCA and LLOCA being reduced in the consequences is the heat removal capabilities of the experimental facilities cooler" (BNL 1993b). This says that only 10 percent of the large LOCA contribution at 40 MW should be counted as full-scale core melt, and only 19 percent of that at 60 MW; the difference in large-scale core damage induced by large LOCA due to power level is therefore (0.19 x $2.06 \times 10^{-4}$) - (0.1 x $1.35 \times 10^{-4}$), or $0.26 \times 10^{-5}$/yr, rather than the $0.71 \times 10^{-5}$/yr that one would calculate as the difference between power levels in the frequency of any core damage at all from large LOCA.

The above consideration strongly affects the presentation of results in Section 7 of *Levels 2 and 3 Internal Events PRA for the High Flux Beam Reactor*. Table 7-4 of *Levels 2 and 3 Internal Events PRA for the High Flux Beam Reactor* shows that complete core damage at 40 MW produces 4 to 6 LCFs, and complete core damage at 60 MW produces 6 to 8 LCFs (BNL 1993b). Based on this, it is reasonable to say that these are characteristic consequence magnitudes for major releases into an unbreached confinement. In contrast, early core damage alone produces very few LCFs, because core damage is arrested, changing the release characteristics considerably. But early core damage is apparently included when the characteristics of the "average" core damage accident are presented in Table 7-1 (2 LCFs per average accident at 40 MW) and Table 7-2 (3.6 LCFs per average accident) (BNL 1993b). Inclusion of early core damage reduces the consequences of the "average" core damage event significantly.

One final bit of perspective from the PRA is afforded by the sensitivity study presented in Table 4-8 of the PRA (BNL 1990a), in which a CDF of $6.4 \times 10^{-6}$ is quoted for large LOCA at a power level low enough for flow reversal not to be an issue. At this power level, according to the Level 1 PRA, there is a success path for large LOCA: injection of poison water followed by light water makeup. This success path is responsible for the difference between the initiating event frequency ($2.1 \times 10^{-4}$) and the CDF ($6.4 \times 10^{-6}$). The Level 2 PRA (BNL 1993b) did not consider this success path; it analyzed the fraction of large LOCA frequency in which core damage was assumed to occur for lack of forced flow and included the potential for this damage to be stabilized short of full core melt.

If it were established that there is no break size for which flow reversal is an issue, the discussion supporting Table 4-8 of BNL 1990a suggests that one would quantify large LOCA CDF at roughly the low-power frequency ($6.4x10^{-6}$) for all power levels, without credit for the EFC (there would be some dependence on power level because of time available to inject makeup coolant). Moreover, according to the Level 2 PRA (BNL 1993b), the EFC can mitigate large LOCA at 40 MW even without operator action, which drives the frequency of this sequence down to the $10^{-7}$ range for 40 MW. At 60 MW, operator action would be needed to drive the frequency to the $10^{-7}$ range, because at that power level, the EFC does not mitigate the accident unless the operator takes action to increase flow through the system. Because of operator involvement in both the makeup success path and the EFC success path, independence of the failures of the two success paths cannot be taken for granted.

In order to establish what break size would lead to minor core damage and to establish the flow rate at which the EFC could arrest core damage, the SAR model for large LOCA was extended to larger break sizes, and the breaks leading to early core damage were examined further to confirm whether the EFC could be expected to arrest core damage (Palmrose 1999). The following results were obtained.

- Breaks smaller than 33 cm (13 in) do not lead to early core damage. Breaks larger than 33 cm (13 in) appear to lead to early core damage.

- If the EFC is operating at 125 lpm (33 gpm) when a break greater than 33 cm (13 in) occurs, minor core damage occurs, but the EFC then stabilizes the core. At 45 lpm (12 gpm) (the normal operating flow rate prior to reanalysis), the EFC does not stabilize the core after a break this large.

In light of the above, the HFBR will operate the EFC at 125 lpm (33 gpm). This means that essentially all large LOCA sequences will lead either to no core damage, or to minor core damage arrested by the EFC. "Major" core damage would not occur unless additional failures occurred beyond those analyzed in the Level 2 PRA (BNL 1993b). These additional failures are considered extremely unlikely; BNL 1993b treated EFC cooling as a stable end state, considering additional failures to be probabilistically insignificant.

As a result of these findings and changes, the modeling of large LOCA is as follows. This explanation is summarized on Table C.5-10

**60 MW Operation:**

The large LOCA quantification in the PRA leads to an initiator frequency estimate of $6.5x10^{-5}$ per year for breaks greater than 33 cm (13 in). It is reiterated that per ANS/ANSI guidance, this break is considered beyond design basis for a facility of this type. At 60 MW, these breaks cause minor core damage, but the situation is then stabilized by the EFC, which will now always operate at 125 lpm (33 gpm). This end state was modeled in the Level 2 PRA as a stable end state with a small offsite release; that source term for radiological release is the basis of the model applied here to characterize the consequences of this event (22 mrem to the MEI). Where the present treatment departs from the Level 2 PRA is in the frequency of breaks leading to this state (reduced, because fewer breaks are now deemed to lead to this condition), and the level of assurance that the EFC will stabilize the situation (now much higher, as a result of more analysis of the EFC's cooling effect, and the new practice of always running the EFC at the higher flowrate).

It is reiterated that the estimate of break frequency used here was considered "conservative" by the PRA analysts themselves. However, reanalysis of it has not been undertaken.

Breaks greater than 33 cm (13 in) lead to an increase in dose rates to operators inside the confinement. BNL 1993b furnished an upper bound on the dose of 2.6 rem incurred as a result of operator actions needed to increase EFC flow and perform other operations after minor core damage at 60 MW. This dose arises essentially because radionuclides released from the damaged fuel are transported to unshielded portions of the system. (Note: Action to increase EFC flow is no longer needed because the EFC will already be operating at the higher flow rate. This dose is well above normal, but not prohibitive of operator action.)

**30 MW Operation:**

The 33 cm (13 in) break can also be postulated for 30 MW operation, and would have the same frequency, but the core will not suffer damage, and the releases will therefore not be off-normal.

***Table C.5-10 Reanalysis of "Large" LOCA Accident Sequences at 60 MW***

| | Thermal/Hydraulic Analysis | Credit for Experimental Facilities Cooler (EFC) | Assessed Frequency of "Minor Core Damage Only" Initiated By Large LOCA[2] | Assessed Frequency of "Major Core Damage" Initiated By Large LOCA |
|---|---|---|---|---|
| Level 1 PRA [BNL 1990a,b] | Based on analysis cited in 1990: All breaks in this range (> 2.06 cm) were assumed to go at least to "minor" core damage due to loss of forced flow | None. It was assumed that after "minor" core damage, there is no way to prevent "major" core damage | None. All breaks in this range (> 2.06 cm) assumed to go to "major" | $2.1 \times 10^{-4}$ /yr (Frequency of Breaks > 2.06 cm ) |
| Level 2 PRA [BNL 1993] | Based on analysis cited in 1993: All breaks in this range (> 2.06 cm) were assumed go to "minor" core damage (i.e., same assumption in this area as BNL 1990a) | (a) EFC flow rate, initially low, needs to be increased after the LOCA occurs; 90% chance operator will do this. (b) 90% chance that major core damage will be prevented if EFC runs at increased flow. | $1.7 \times 10^{-4}$ /yr (81% of All Breaks > 2.06 cm ) | $3.8 \times 10^{-5}$ /yr (Remainder, i.e., 19%, of All Breaks > 2.06 cm ) |
| Present Analysis | Supplementary Analysis [Palmrose 1999]: Only breaks > 33 cm cause "minor" core damage due to loss of forced flow | (a) EFC will permanently run at increased flow. (b) EFC running at increased flow will prevent major core damage [Palmrose 1999] | $6.5 \times 10^{-5}$ /yr Assessed Frequency Of Breaks > 33 cm[1] | Residual (less than $10^{-7}$/yr) (Additional, unlikely failures would be needed to get to major core damage) |

Notes.

1. The Level 1 PRA [BNL 1990 a,b] quantified the frequency of all breaks > 2.06 cm and itemized contributions from different break sizes. The frequency quoted here for breaks > 33 cm is obtained from BNL 1990b by counting contributions only for breaks > 33 cm.
2. This is the end state analyzed in the Level 2 PRA [BNL 1993b] for successful EFC cooling following minor core damage. Refer to Fig. 4.7 of BNL 1993.

### *C.5.1.1.2.2 Loss of Offsite Power (LOOP)*

LOOP is within the design basis, and LOOP events have occurred at the HFBR (Schmidt 1998 provides recent information). Only a small fraction of LOOP initiators are expected to lead to core damage. This small fraction arises because a LOOP assumes that power fails, cooling fails, inventory makeup fails, and the power outage lasts so long that coolant inventory is boiled off before power is restored.

For specificity, the following discussion relates to the largest contributor to LOOP CDF as determined in the Level 1 PRA (See Table 5-6 of BNL 1990a) and the most likely path through the Level 2 event trees (Fig. 4.19 of BNL 1993b for 40 MW, and Fig. 4.9 of BNL 1993b for 60 MW).

**Scenario Frequency:** No significant difference in LOOP initiating event frequency is believed to exist for different operating power levels.

**Scenario Evolution:** For the dominant LOOP sequence in the PRA, there is no cooling or addition of coolant (makeup), coolant boils off as a result of heat production, and eventually the fuel releases its radionuclides. It takes longer for this to happen if the reactor had been operating at lower power (30 MW or 40 MW vs. 60 MW). Human operator actions are modeled as having different failure probabilities for different time frames available for action, and the available time frame for action in LOOP leads to some differences between CDFs assessed at different power levels. Thus, the frequency for the LOOP event (initiating event plus follow-on actions) is greater at 60 MW than at 30 MW (See Tables C.5–15 and C.5–16).

**Consequences:** The scenario evolves more slowly if the reactor had been operating at lower power. Therefore, at lower power, there is more time for decay of radionuclides and the release occurs later. Confinement does not fail; performance of filtration is partially degraded by steam and humidity, but there is some filtration by the HEPA and the charcoal filters. The initial inventory is slightly different for different power levels, and the decay time from shutdown to release is different.

In the scenario discussed here, the results of the original PRA are:

For 40 MW: If complete core damage occurs given LOOP, then 90 percent of the time, the release characteristics cause 4 LCFs offsite, and 0.1 LCFs onsite, at a frequency of $1.5x10^{-5}$/yr. (Note: This is not the totality of all LOOP accidents, just the most frequent release category.)

For 60 MW: If complete core damage occurs given LOOP, then 90 percent of the time, the release characteristics cause 6 LCFs offsite, and 0.2 LCFs onsite, at a frequency of $4.8 \times 10^{-5}$/yr. (Note: This is not the totality of all LOOP accidents, just the most frequent release category.)

As noted before, the consequences from this kind of scenario are essentially determined by the considerations that the release into confinement is significant, but filtration, while not completely successful, works well enough to limit consequences to these levels.

The alternatives included in this report are 60 MW operation and 30 MW operation, but not 40 MW operation. It is observed that the consequences reported in the PRA for these LOOP accidents seem to scale with power level. This is not surprising, but neither is it necessarily due to a simple scaling of fission product inventory with power level. It could also be due to differences in accident timing. From Table 4-8 of BNL 1990a, it can be seen that although the frequency of LOOP core damage at 60 MW is greater than that at 40 MW, the CDF at 40 MW and below 40 MW are the same.

**Effect of Facility Improvements:** Makeup water sources have been added since the PRA modeling was completed in 1994. Since the dominant sequences involve failure of makeup water sources, the addition of an additional source of water would be expected to reduce the CDF. Moreover, as mentioned in the PRA (BNL 1990b) the human error probabilities are based on conservative time windows and on a screening human reliability analysis.

A partial reanalysis or updating of the LOOP initiated accident was therefore undertaken to estimate the impact of these design improvements and other factors on the CDF (Schmidt 1998). This reanalysis consisted of adjusting the results of the original Level 1 PRA (BNL 1990a, BNL 1990b) to include the effects of:

- The addition of the Secondary Poison Water Addition Method (SPAM) system and the Long Term Light Water Makeup (LWM) system (BNL 1996b)
- A reanalysis of the human error associated with Poison Water System (PWS) initiation and long term makeup
- Taking credit for recovery of long term water sources on recovery of offsite power
- Current estimates of time available to achieve makeup
- Elimination of the assumption that less than 3 minutes of forced flow following shutdown would lead to core damage
- Updated LOOP frequencies to include site data since the PRA and the latest generic data
- The HFBR operating cycle

Rather than starting from the beginning, the reanalysis was closely based on the original PRA results. Contributions to LOOP CDF were separated according to how the above-listed factors would affect them; the appropriate factors were applied to each contribution; and the modified contributions were re-added to obtain the new estimate of LOOP CDF.

The original and revised LOOP CDF results are shown in Table C.5-11.

In order to prevent confusion, Table C.5-11 provides updated results including and excluding ATWS, since ATWS sequences initiated by LOOP are unique and significantly different from the lack of makeup sequences initiated by LOOP. The minor increase in LOOP ATWS CDF at the lower power level is a result of the longer operating cycle permitted by the lower power level. ATWS contributions have not been reassessed here in detail, and are not included as part of the LOOP scenario in the comparison of alternatives.

The updated CDF for all LOOP initiated accidents is significantly reduced from the original value, primarily because of the impact of the SPAM/LWM system and the more realistic human error probabilities. The effect of power level on the total LOOP CDF is, however, less for the updated analysis than for the original analysis. In the original analysis, the LOOP CDF was dominated by loss of makeup sequences (which are reduced at lower operating power), with ATWS sequences (whose frequency was not assessed as being affected by power level) being a very minor contributor. The updated result shows a significant reduction in the frequency of loss of makeup sequences at 60 MW, hence, further reductions at lower power lead to only minimal reduction in total LOOP CDF. If ATWS sequences are excluded, the power dependence is significant. Although the frequency of ATWS sequences was not

***Table C.5-11. Comparison of Original and Revised Loss of Offsite Power Core Damage Frequency Results***

| Power Level | Original PRA Including ATWS | Updated Analysis Including ATWS | ATWS | Excluding ATWS |
|---|---|---|---|---|
| 60 MW | $5.3 \times 10^{-5}$ | $1.1 \times 10^{-6}$ | $8.0 \times 10^{-7}$ | $2.6 \times 10^{-7}$ |
| 30 MW | $1.6 \times 10^{-5}$ | $9.4 \times 10^{-7}$ | $8.5 \times 10^{-7}$ | $8.6 \times 10^{-8}$ |

Source: BNL 1990a, Schmidt 1998.

changed significantly in the update, they are now relatively more significant, because other contributions were reduced. This does not mean that the comparison of alternatives should now be based on ATWS sequences; in order to justify comparing alternatives on the basis of ATWS, a careful reconsideration of their uncertainties and sensitivities would need to be undertaken.

The updated results are based only on a partial revision to the PRA. This revision did not include revising the event trees and fault trees nor a comprehensive requantification of all accident sequences. The adjustments were applied conservatively and other corrections and/or updates are warranted. However, the results for the sequences examined are considered to be valid for the purposes of evaluating the environmental impacts of the alternatives in this EIS.

### *C.5.1.1.2.3 Beam Tube Rupture*

The potential BTR event modeled in the PRA is a very severe event. It is a very large postulated leak in a location that makes it more severe in some respects than it would be if a break of similar size occurred elsewhere in the primary system. The location of the leak creates the potential to uncover the core more quickly than in most other accidents. Also, when the PRA was being developed, the assumed size of the leak was assumed to lead to "early core damage" (premature loss of forced flow causing overheating of the core). The flow reversal issue for BTR is closely related to the flow reversal issue for a large LOCA. As discussed in Section C.5.1.1.2.1, recent analysis substantially changes the understanding of flow reversal from the understanding that was current when the PRA was performed. This suggests that the modeling of the evolution of BTR accidents should change substantially, possibly leading to a reduced estimate of CDF even if the frequency of the initiating event were not changed. At present, this has not been done.

In the PRA, the BTR event is modeled as leading to major core damage. As noted above, all internal event accident sequences that lead to essentially complete core damage have generally similar consequences (Table 7-4 of BNL 1993b). Some differences arise because of timing, but grossly, the release occurs into an intact confinement with filtration partially degraded as a result of steam and humidity.

The modeling of the frequency of the accident — the frequency of BTR itself — is subject to considerable uncertainty and some debatability. The issues are summarized later in this section. For purposes of this impact assessment, the medium-consequence accident presented for comparison of alternatives will be LOOP rather than BTR. For fundamental reasons mentioned in the preceding paragraph, the consequences of a LOOP accident are grossly similar to those of a BTR, and both appear to scale similarly with operating power level. The frequency of core damage due to these events is also assessed in the PRA as being grossly similar, although the frequency of LOOP-induced core damage varies with operating power level while that of BTR does not (at least, not between 60 MW and 40 MW in the PRA), and the frequency of the LOOP sequences is perhaps more certain.

A question has been raised whether there are safety implications associated with the Resume Operation and Enhance Facility Alternative, beyond the hazards associated with the operation of revesseling itself. There is no reason to expect that anything would get worse by design; no system failure probabilities would increase, nor would any system failure probabilities necessarily decrease. The question is simply whether renewal of the vessel and the beam tubes has a safety benefit.

The existing PRA quantifies some risk associated with the possibility of a BTR accident. The frequency and consequences associated with BTR in the PRA are significant *compared to other PRA findings*, but, like other contributors, generate only small average (frequency-weighted) consequences. Even though the contribution to risk is small on an absolute basis, it is natural to ask whether it is driven by degradation of beam tubes, and if so, whether enhancing the facility would improve the metallurgical condition of the vessel and the beam tubes and thereby affect the frequency of the rupture event. Based on a reading of the original PRA, it appears that the PRA's assessment was not driven by a perception of a significant degradation in materials properties. This will be addressed below. If this were true, then replacement would not necessarily be viewed as a net improvement.

Additionally, it seems likely that in light of today's information base, the existing PRA somewhat overstates the risk contribution from BTR. Overstating the current risks would tend to overstate the safety benefits of enhancement, if, in fact, enhancement has any safety benefit.

Quantifying the expected frequency of a large rupture event is not trivial. The original PRA estimate was based on expert elicitation. Today, a similar exercise would make use of more data than was available for the original PRA, because, for example, survey work and additional non-destructive examination have been done on the beam tubes. Alternatively, an investment in more explicit physical modeling could be considered. The comments made here regarding what the results of reanalysis might be are not intended to preempt or substitute for reanalysis, but are essentially engineering judgment, offered to afford some perspective on the meaning of the existing results and issues that might be considered in a new analysis.

**Summary Discussion of Quantification of the Frequency of BTR:** The contribution of BTR to risk in the PRA is somewhat different in nature from that of most of the other internal-events contributors. The BTR frequency is a mean value that was strongly influenced by the large uncertainty built into the underlying frequency distribution that was used to determine the mean. The expert on whose input the distribution was based apparently believed in a lower number, but allowed that the frequency might be higher, and it is the averaging over these possibilities that led to the number used in the PRA (BNL 1990b). That is, the apparent significance of BTR in the risk profile is driven by the uncertainty surrounding the frequency, rather than by a positive belief that the frequency is near the upper limit of the assigned distribution. Formally, this point applies to all of the contributors in the PRA, but its effect on BTR is much greater than for other contributors because the assessed uncertainty is so high for this particular event. (See below for a more specific discussion.) The "mean values" of the other contributors are closer to the peaks of their distributions.

The discussion in the PRA indicates that the frequency distribution for BTR was not based on any observation of significant radiation damage in the existing tubes, but was simply the metallurgical expert's acknowledgment that rupture is not ruled out on grounds of physical impossibility. Therefore, it is not clear that simply replacing existing tubes would justify a revised estimate to that expert.

Note, however, that the expert's median estimate was already nearly two orders of magnitude lower than the mean value derived from his input by the PRA analysts. Since the mean is driven by uncertainty, any reduction in uncertainty would translate into a reduction of the mean risk. Revesseling might provide a basis for reduced uncertainty.

Even without revesseling, more recent work creates a basis for a revised estimate of the likelihood of BTR. For one thing, the condition of the tubes has been extensively surveyed. The conclusion of the surveys supports the integrity of the current beam tubes' pressure boundary. This provides evidence that was not available when the PRA was done, and would lead to a revised estimate of the mean frequency of BTR, just by reducing uncertainty.

Whatever the current risk is, at some point when major components have been subjected to enough radiation and their properties finally begin to change, further operation would begin to introduce additional risks that revesseling would ameliorate.

From the above discussion, the following conclusions are presented:

- The accident contributor in the PRA that may be affected most significantly by revesseling is the BTR contribution.

- The key to reducing the contribution from BTR appears to be reducing the uncertainty in the underlying probability-of-frequency distribution, or, to put it another way, ruling out the upper bound of the frequency spectrum.

- If reanalysis of the BTR frequency were undertaken based on recent surveys of the tubes, and on whatever other work may have been done since the preparation of the PRA, the assessed risk contribution might be reduced without revesseling. (This comment assumes that new information would tend to confirm the gist of the previous assessment, and simply reduce the uncertainty range.)

- If today's risk contribution from BTR were not so reduced, revesseling might furnish some basis for reducing it, if in some way revesseling generated new information that could be invoked to reduce uncertainty.

- Since the current estimate was apparently driven by uncertainty rather than by a perception of significant radiation damage, it is not clear that revesseling would eliminate the current contribution unless it somehow reduced uncertainty.

- The analysis has not been done, but the information cited suggests that with uncertainty assessed in a consistent fashion based on today's information, the absolute difference between the frequency of BTR for the Resume Operation and Enhance Facility Alternative and for the other alternatives would be small, and that if uncertainty could be reduced based on current information, both cases would yield smaller risk contributions than are reflected in the current PRA.

- After many additional decades of operation, and additional radiation damage, the prospect of BTR will be less remote (see discussion below); any additional risk contribution assessed at that time would be ameliorated by revesseling.

- The events causing and following a BTR needs to be reassessed in light of the improved understanding of the flow reversal issue.

**Specifics of Quantification of the Frequency of BTR:** The relative significance of the contribution from BTR is such that closer examination of its basis is warranted. The event is outside the envelope of events considered in the SAR. It is examined in the PRA, along with other events that are rare, but create special problems for mitigating systems.

Figure 4.1 from BNL 1990a shows uncertainty bands for frequencies of major categories of contributors to core damage. The uncertainty band for BTR is seen on this figure to span four orders of magnitude, a much broader range than that spanned by other contributors. The mean of the assumed underlying distribution, just over $1x10^{-4}$ per year, is the frequency quoted in the results section of the PRA for major core damage due to BTR.

The source of this uncertainty band and the form of the assumed underlying distribution are presented in BNL 1990b. Figure 5.8 of that reference shows the actual distribution; page 4-4 of that reference summarizes how it was derived. Essentially, in an expert elicitation process, one expert in materials behavior provided "upper bound" (taken by the PRA analysts to be a $95^{th}$ percentile), median ($50^{th}$ percentile), and lower bound (taken by the PRA analysts to be a $5^{th}$ percentile) estimates of $2x10^{-4}$, $2x10^{-6}$, and $2x10^{-8}$, respectively, on the assumption that the underlying distribution is log-normal, the mean is then derived as $1.01x10^{-4}$. The "mean" of this distribution does not therefore correspond to anything like a "best guess," which some would identify with the peak in the distribution, but rather reflects how the mathematical process of averaging a broad log-normal distribution places the mean near the high end. The expert's original memo, provided as an attachment in the PRA (BNL 1990b), says:

> Since I believe leak before break will apply to HFBR tubes, based on the French experience and other reasons cited in my earlier memo, I think any failure rate > $10^{-4}$ per year would be unreasonably high. A lower bound $10^{-7}$ is probably a reasonable guess, with a median ~$10^{-5}$ (which is consistent with the HFIR [High Flux Isotope Reactor] large LOCA assumption of $1.4x10^{-5}$ for a median). Given the low pressure in HFBR the compressive stresses on the tubes and their increasing strength with time, we could or should decrease this guess by at least another factor of 10, using a median of ~$2x10^{-6}$/year, upper bound $2x10^{-4}$, lower bound $2x10^{-8(?)}$.

That is, the peak in the distribution (Fig. 5.8 of BNL 1990b) reflects what we might call the expert's best guess (that is, $2x10^{-6}$/year); the "mean," located much higher, is derived from assuming a log-normal distribution that goes through the upper and lower bounds.

On the further assumption of unmitigability of the postulated event, this "mean" becomes the average (over uncertainty) of the frequency of BTR-induced core damage.

Note that in principle, the frequency of BTR may be age-dependent: at some point, consideration of aging damage would increase the assessed likelihood of an accident. Earlier in the same memo, the expert commented that he would "start worrying when ductilities dropped below 2% … The equation predicts this level at … three times the present [that is, the 1989] dosage." This suggests that barring some future information that changes the current picture, based on the expert's considered opinion, the beam tubes could provide the opportunity for some decades of additional safe operation.

If a completely new set of beam tubes is installed in the future, the frequency estimate derived above should be re-evaluated. Because the expert actually cited increasing beam tube strength with time in his argument for a best estimate of ~$2x10^{-6}$, we are not justified in assuming that the lower number would apply to the newer component. On the other hand, perhaps the uncertainty that drives the "mean" as explained above could be reduced for new components, if information that reduced the uncertainties implicit in the expert's recommendation became available.

Any attempt to compare the accident picture for different alternatives would need to be done on the basis of rather carefully benchmarked uncertainty estimates. Keeping an uncertainty-driven number for the no-enhancement cases, and generating an estimate having artificially reduced uncertainty for the Resume Operation and Enhance Facility Alternative, could overestimate the safety benefit of enhancement. The proper comparison between enhancement and no enhancement should ideally be based on quantification of the present system in light of current, irreducible uncertainties, and quantification of the as-enhanced situation, in light of probable uncertainties that apply for that case. The analysis has not been done, but the information cited suggests that the absolute difference between the two cases would be small, and that both cases would yield smaller risk contributions than are reflected in the current PRA.

### *C.5.1.1.2.4 Severe Wind Event*

The possible SWT event is modeled as causing a LOOP, thereafter breaching the confinement structure, and also causing failure of coolant makeup. Failure of confinement and failure of coolant makeup are assumed to be caused by wind-driven projectile damage, that is, by heavy objects driven by the wind through the confinement structure and into various systems. Apart from initiating the event, the evolution of this accident resembles the LOOP accident described above, in that there is a slow boiloff of coolant followed by a release of radionuclides into the confinement structure. However, in one key respect, this accident is different: the confinement structure is breached and the estimated release to the environment would therefore be much larger.

Facility upgrades since the PRA was performed may potentially lower the assessed frequency of this event. Seismically qualified sources of poison water and light-water makeup have been added to the plant. Because there are more sources of coolant than before, it is less likely than before that all sources of coolant would be disabled by the postulated wind-driven projectile. To be consistent in intent with the PRA model for the original configuration, a new model would require the wind-driven objects not only to destroy the makeup systems that existed at the time of the original model, but also to incapacitate the new, seismically qualified sources. If the joint probabilities of all these events were shown to be reduced from the probability of destroying the original sources, then the assessed accident frequency would be reduced.

As modeled in the PRA, the frequency of the event is independent of power level. For the more likely radiological release (no core-concrete interactions involved), the following results are obtained:

At 40 MW: 75 LCFs offsite, 1.9 LCFs onsite; no prompt fatalities offsite, $8.2 \times 10^{-3}$ prompt fatalities onsite. This is not a prediction of a fraction of a death, but means that there is a small chance that particular weather conditions could disperse the release in such a way as to lead to a prompt fatality.

At 60 MW: 108 LCFs offsite, 2.3 LCFs onsite; $1.7 \times 10^{-5}$ prompt fatalities offsite, $2.9 \times 10^{-2}$ prompt fatalities onsite. Again, this is not a prediction of a fraction of a death, but means that there is a small chance that particular weather conditions could disperse the release in such a way as to lead to a prompt fatality.

The very significant rise in the per-accident consequences (compared to the per-accident consequences for LOOP, large LOCA, and BTR) is chiefly a result of the breached confinement. The per-accident consequences are almost the largest shown on Table 7.1-4 of the PRA; this, combined with the relatively high frequency of this accident, causes it to dominate the frequency-weighted consequences for external events.

Because of modifications to the facility, a partial reanalysis or updating of the tornado-initiated accident was undertaken to estimate the impact of the above design improvements and other factors on the core damage frequency (Schmidt 1998). This reanalysis consisted of adjusting the results of the original Level 1 Tornado PRA (BNL 1993b and BNL 1994) to include the effects of:

- The addition of the SPAM system and the Long Term LWM system (BNL 1996b) accounting for projectile damage probability
- A reanalysis of the human error associated with PWS initiation and long term makeup
- A revision/correction to the projectile penetration and damage probability to account for double counting of damage to adjacent components and elimination of projectile damage to system that fails safe
- Current estimates of time available to achieve makeup
- Elimination of the requirement for 3 minutes of forced flow following shutdown
- Revised tornado hit frequency to be consistent with the basis for projectile probabilities and observed experience
- Limited potential for recovery of makeup by offsite personnel if control room personnel are incapacitated
- The HFBR operating cycle

The reanalysis was done by adjusting the available results for the 10 dominant core damage sequences as provided in the Tornado PRA report (BNL 1994) to account for the impact of these factors. Because available information only presented the most likely combinations of failures contributing to core damage frequency, the adjustments were applied only to those contributions. The remaining contributions were unchanged or adjusted only by global factors that are known to affect the frequency of all combinations of failures in a similar way (such as tornado frequency). The revision did not include repair of equipment damaged by the tornado, and thus remains conservative, particularly at 30 MW where over a day is available to make such repairs.

***Table C.5-12. Comparison of Original and Revised Severe Wind/Tornado Accident Core Damage Frequency Results***

| Power Level | Original PRA | Updated Analysis |
|---|---|---|
| 60 MW | $4.0x10^{-5}$ | $8.7x10^{-7}$ |
| 30 MW[a] | $4.0x10^{-5}$ | $7.9x10^{-7}$ |

[a] The original PRA value for 30 MW was inferred, not calculated.

Source: BNL 1993b, BNL 1994, Schmidt 1998

The original and revised tornado initiated CDFs are shown in Table C.5-12.

The revised analysis is seen to result in a very significant reduction in CDF. A large part of this reduction is due to the use of a frequency for Fujita F-scale classification F3 tornado (250 kmh to 330 kmh [158 mph to 206 mph]) as the frequency of the tornado that generates the damaging projectiles rather than that for an F1 tornado (73 mph to 112 mph) as used in the original analysis. The projectile velocities and impact probabilities were based on winds uniformly distributed over the range of 240 kmh to 480 kmh (150 mph to 300 mph) (BNL 1993b). The use of the higher frequency associated with wind speeds lower than the projectile analysis results in excessively conservative results. The benefits of the SPAM/LWM and of various corrections in the analysis account for the remainder of the reduction.

The dominant contributor to the CDF is failure of all cooling due to standby propane generator failure and projectile damage to the PWS and

SPAM, or failure of offsite personnel to provide makeup after control room personnel are incapacitated by a projectile. The minor change in CDF at different power levels is due to the contribution of projectile damage to equipment because no credit is taken for the repair of damaged equipment. Because of this, the frequency for the SWT event (initiating event plus follow-on actions) is greater at 60 MW than at 30 MW.

### *C.5.1.1.2.5 Fuel Handling Accident*

In the postulated FHA scenario, insufficient cooling of a spent fuel element during a discharging operation would lead to fuel melt and the release of fission products to an intact confinement.

This is an infrequent incident, stated in the SAR to result in the largest computed offsite dose consequences of all credible HFBR accidents (BNL 1998). As analyzed in the SAR, the thyroid doses are limiting; depending on time since shutdown, the computed offsite thyroid dose from this accident approaches 5 rem, a consequence level beyond which offsite emergency response would be required. Therefore, the waiting period after shutdown before the discharging operation is determined by the requirement to keep thyroid doses below 5 rem (the radioiodines responsible for this dose are allowed to decay before the fuel is handled, in order to limit the consequences of this event).

Accident frequency is assessed based on a probability per fuel element discharged, multiplied by an assumed number of elements discharged per year. The frequency quoted here for power levels other than 60 MW has been scaled from the 60 MW result, based on the assumption that the 60 MW result corresponds to half-core replacement and that the 40 MW operating cycle would be 24 days operating, followed by replacement of a quarter of the core. The PRA did not quantify the relative number of elements handled at different power levels.

### *C.5.1.1.2.6 $D_2O$ Release Accident*

In this postulated event, a leak would develop in the primary heat exchanger, permitting tritium-bearing $D_2O$ to leak from the primary side to the secondary side. The core would not be involved in this event, but since the primary coolant contains tritium, the event would have the potential to release tritium to the environment. The event is assessed "infrequent," meaning that it has a frequency between $10^{-3}$ per year and $10^{-1}$ per year.

The magnitude of the consequences of the event is determined by how long it takes to detect the leak. Three methods of detection are described in Table 3.4.23a of the SAR, "Sensitivity of Detection Methods for Primary-to-Secondary Leaks – Reactor Operating." The consequences vary according to the leak size and the assumption made about whether in-line tritium monitors would be in service.

The accident analysis consequences for this event indicate that the involved worker could experience a dose rate on the order of 1.3 mrem per hour if the in-line tritium monitors are out of service. These doses are not significant for the few hours over which exposure could reasonably be expected to occur. Doses to onsite noninvolved workers and airborne offsite doses would be insignificant.

The potential radiological significance of this event does not stem from any airborne source term, but rather from the potential of releasing contaminated water to the environment as a result of not detecting the leak for some time, and then, after discovering the leak, not remediating any discharge that had occurred. For the out-of-service assumption cited above for in-line tritium monitors, and a leak rate just below the alarm point of the secondary gamma monitors (4 $cm^3$/sec of primary coolant), the

SAR states that the secondary cooling water could reach a tritium concentration of 0.41 $\mu Ci/cm^3$.

As shown on Table 3.4.23a of the SAR, active monitoring for this particular leak is done in several ways by independent means. No scenario has been analyzed in the SAR in which monitoring fails for an extended period and remediation is not undertaken thereafter. Involved worker health impacts are extremely small for the scenario in which the release is detected and mitigated; there would be no public heath impact.

#### *C.5.1.1.2.7 Experimental Facilities Accidents*

Part of the mission of the HFBR is to perform in-core irradiations that generate radionuclides for medical and research applications, as well as other experiments that may involve small amounts of radioactive material. The in-core irradiation experiments may involve significant amounts of potentially hazardous radioactive material. The other experiments typically involve much smaller amounts of radioactive material. In the context of a discussion of a spectrum of accidents from the HFBR, it seems warranted to consider accidents involving these facilities. Unfortunately, the variety of possible irradiations and the potential different experiments make it difficult to explicitly model a representative frequency and consequence for this category of possible events. However, some conclusions can be drawn from analyses described in previous subsections, and from operating history.

In examining the potential for environmental impacts due to releases from accidents associated with these experiments, useful perspective is afforded by the analysis of the FHA described above. In that event, a fuel element would be essentially destroyed, but its release, driven by fission product afterheat, would be into an intact confinement with fully operational filtration. The computed radiological consequences are minimal, as discussed above, even on the basis of a conservative consequence model.

In considering an extrapolation from this event to a release somehow arising from irradiated material, the first point to be made is that the hazard associated with irradiations would be significantly lower than for a fuel element. Secondly, the energies driving an airborne release would be much lower. The fuel element would be generating a large amount of heat, which is why its airborne release occurs; irradiated material typically generates much less. Finally, the HFBR confinement structure is a highly effective barrier to this kind of airborne release.

The situation would be slightly different for other experiments. The activity present in experiments other than irradiations is typically significantly less than in irradiations. On the other hand, the potential exists for driving the material into the HFBR confinement atmosphere by an event such as a fire in the experimental apparatus. An event of this general type occurred at the TRISTAN facility (Davis 1994). In that event, several staff members inside the HFBR confinement were contaminated slightly, but releases outside the HFBR confinement were minimal, as would be expected. The TRISTAN facility has been removed.

After that event, it was determined that review of the TRISTAN experiment had been inadequate. As a result, procedures for reviewing experiments with a view toward preventing such events were considerably enhanced. Before the new procedures were in place, it would have been warranted to assign a frequency between "moderate" and "infrequent," or, $10^{-1}$ per year; after enhanced review procedures were adopted, one would assign a frequency reduced from this by orders of magnitude. The consequences of representative events in this event class should be significantly less than 1 mrem for people outside the HFBR confinement structure.

It is conceivable that larger consequences might result from radiological releases to the confinement atmosphere having some chemical characteristic that reduced the efficiency of the filtration. This would be assessed on a case-specific basis.

### *C.5.1.1.2.8 Aircraft Crash Analysis*

The potential for an aircraft impact into the HFBR was analyzed in the 1993 *Aircraft Impact Analysis for the HFBR* (BNL 1993a). The results of these analyses were used in the HFBR PRA to evaluate the risks of external events. The PRA concluded that an airplane crash scenario did not significantly contribute to the HFBR risk profile. Therefore, it was not selected for comparison purposes in this EIS. However, during public scoping, some stakeholders raised a concern about a potential aircraft crash into the HFBR site and HFBR building. To address this concern an overview of the PRA results including the probability of a crash and the possible consequences are discussed below. Where appropriate, comments on conservatism and analysis uncertainty are included.

The PRA analysis considered three sources of aircraft traffic: local airways, general aviation traffic from Brookhaven Airport, and the Grumman Calverton site. Three types of aircraft were considered: military, commercial, and small aircraft (including both twin and single engine planes). Note that the PRA considered the traffic from the Grumman Calverton site, however, Grumman has since closed that facility. Therefore, the potential for a military plane crash has been significantly reduced.

The PRA analysis considered the likelihood of an aircraft impact to HFBR structures based on their relative size. For example, an aircraft crash into the HFBR site was assumed to occur at the HFBR building (55 percent), the HFBR stack (26 percent), the cooling towers and associated pump house (18 percent), and the D.C. transformer building (less than 1 percent). The frequency of an aircraft crash to the HFBR site and HFBR building are shown in Table C.5-13.

***Table C.5-13. Frequency of Aircraft Crash***

| **Aircraft Type** | **HFBR Site (per year)** | **HFBR Building (per year)** |
|---|---|---|
| Single Engine | $2.81\text{x}10^{-5}$ | $1.57\text{x}10^{-5}$ |
| Twin Engine | $5.06\text{x}10^{-6}$ | $2.78\text{x}10^{-6}$ |
| Military | $1.04\text{x}10^{-7}$ | $4.35\text{x}10^{-8}$ |
| Commercial | $2.11\text{x}10^{-6}$ | $9.19\text{x}10^{-7}$ |

Source: BNL 1993a

Note: Military flights (and risk) are overstated, as Grumman's Calverton site is closed.

Consequence determinations were made based on the size and speed of the respective aircraft and the potential for a collision to penetrate the HFBR confinement dome resulting in facility and equipment damage.

The Level 2 and 3 External Events PRA calculates the frequency of a single engine aircraft crashing into the HFBR building as $1.57\text{x}10^{-5}$/yr with an insignificant probability ($8.0\text{x}10^{-6}$ per crash) of penetrating the HFBR dome and causing any damage to the reactor or reactor systems. A single engine plane impact would be insufficient to penetrate the HFBR confinement dome.

The PRA calculates the frequency of a twin engine aircraft crashing into the HFBR building as $2.78 \times 10^{-6}$/yr with a 41 percent chance of penetrating the HFBR confinement dome. If the aircraft does penetrate the dome, it is assumed that core damage occurs as a result of a loss of reactor controls, similar to a LOOP with open confinement scenario. No credit was assumed for personnel and equipment to take action to prevent core damage during the time available for mitigating actions (a minimum of eight hours following an accident during operations at 60 MW).

The PRA calculates the frequency of commercial or military aircraft crashing into the HFBR building as $9.19 \times 10^{-7}$/yr and $4.35 \times 10^{-8}$/yr, respectively. The PRA assumes that core damage with open confinement would occur in these extremely unlikely scenarios.

### C.5.1.1.3 Supplementary Consequence Analyses

In order to support the present decision process, consequence analyses were needed beyond those provided in the PRA. The PRA did not in general quantify consequences to MEIs and the non-involved worker population, and did not provide consequence information on the FHA. The PRA also did not quantify consequences for 30 MW operation, and its estimates of the consequences for 40 MW operation were derived by scaling the core fission product inventories by the ratio of the power levels. For present purposes, additional consequence estimates were required, and calculations were performed to provide them (Wagage 1999). This subsection summarizes the basis for those calculations. The results are provided in Tables C.5-15 and C.5-16.

Like the original PRA consequence analyses, the present calculations were performed using MACCS (SNL 1990a, SNL 1990b, and SNL 1990c). The MACCS computer code uses six input decks as follows:

- Atmos.inp treats the atmospheric transport and dispersion of material and its deposition from the air
- Early.inp models the effect of the accident on the surrounding area during an emergency period, which can last up to one week
- Chronc.inp considers the impact in the period subsequent to the emergency action period, out to infinite time
- Site.inp specifies the population distribution and land use information for the region surrounding the site
- Dose.inp provides dose conversion factors for 60 radionuclides and 12 organs for cloudshine, groundshine integrated for eight hours, groundshine integrated for seven days, groundshine dose rate, internal doses from inhalation for lifetime exposure, and internal doses from ingestion
- Weather.inp provides one year of hourly recordings (a total of 8730) of the wind direction, wind speed, atmospheric stability, and accumulated precipitation

The starting point for the present analyses was the set of input decks previously used by BNL (BNL 1993b, BNL 1994). Of these input decks, site.inp, dose.inp and weather.inp were used without change (and thus the same meteorology was used in calculating the consequences of all accidents), and early.inp and chronc.inp were changed only for selecting the output. Changes were made in the atmos.inp input deck as described below.

For accidents initiated at power (for example, LOCA, SWT, and LOOP), a fraction of the core fission product inventory was assumed to be released. For these accidents the core fission product inventory was obtained from Karol 1998.

The operating cycle and refueling scheme are varied with power level. For 60 MW operation, the end-of-cycle inventory corresponding to the half-core refueling scheme was used; for 30 MW operation, the end-of-cycle inventory corresponding to the quarter-core refueling scheme was used. This is consistent with actual operating practice.

For the FHA, the fission products released through the stack during a FHA were obtained from the HFBR SEG File No. 69 (BNL 1997b). BNL 1997b calculated releases through the stack based on the inventory for a peak fuel element. The HFBR SAR notes that refueling can initiate in seven days after shutdown for 60 MW power operation and in two days for 30 MW power operation (BNL 1998). Therefore, the release quantities were obtained for the FHA at seven days after shutdown for 60 MW power operation and at two days after shutdown for 30 MW power operation. For the FHA the fission products released through the stack were released to the environment without further reductions.

For both the SWT and LOOP accidents, the timing for alarms and releases were changed to reflect the current best estimates (Ports 1998b and Ports 1998c). Also, for all analyzed accidents, no evacuation or relocation of the offsite population was assumed.

Accident sequence 3A of the internal events PRA (BNL 1993b) was assumed to represent the large LOCA accident. Accident sequence 4B of the external events PRA (BNL 1994) was assumed to represent the SWT accident. Accident sequence 3B from the internal events PRA (BNL 1993b) was assumed to represent the LOOP accident.

Non-involved workers were assumed to be located in the area between 0.4 km (0.25 mi) and 1.6 km (1 mi) of the HFBR. The offsite population dose was calculated assuming that the site boundary was at a radius of 1.6 km (1 mi) to 80 km (50 mi) of the HFBR. The non-involved worker population distribution was taken from BNL 1993b. The offsite population distribution was calculated using SECPOP90 (Humphreys 1997). The total non-involved worker population and offsite population used in the analysis were 2,686 and 5,356,270 persons, respectively. The dose to the maximally exposed off-site individual was assessed at 3,000 meters, corresponding to a point on the site boundary north-northeast of the HFBR. The above populations and locations were used consistently for analyzing the consequences of the selected accidents.

The dose-to-risk conversion factors used in the consequence analyses were selected based on the guidance in Section C.2.1.2. A dose-to-risk conversion factor of 0.0005 LCFs per person-rem was used for the public population doses and a factor of 0.0004 LCFs per person-rem was used for the non-involved worker population doses, since for each accident analyzed, almost all, if not all, of the individuals in the population were projected to receive a dose less than 10 rem. A dose-to-risk conversion factor of 0.0005 LCFs per person-rem was used for all MEI doses calculated to be less than 10 rem (that is, all analyzed accidents except the SWT accident). A dose-to-risk conversion factor of 0.001 LCFs per person-rem was used for the SWT MEI dose since this dose was calculated to be greater than 10 rem.

The population LCF calculations performed for accidents initiated at power for the 60 MW case produce results similar to those provided in the PRA (BNL 1993b, BNL 1994). There are minor differences in the results, but these can be attributed to the modified inputs (such as different timing for alarm and release and different core fission product inventories) used in the present analysis. The PRA did not analyze 30 MW, but it analyzed 40 MW; comparison of the present 30 MW results with the PRA's 40 MW results also shows the expected correspondence between the two calculations.

It is reiterated that the PRA did not furnish MEI dose information, and the consequence calculations discussed in this section were done primarily in order to provide a consistent set of MEI doses, not to reflect significant changes in physical models or assumptions.

### C.5.1.2 Analysis Methodology

This section summarizes and applies the results of the previous section to make a final selection of scenarios to serve as the basis for comparison of alternatives.

Owing to improvements in the technical basis for accident sequence analysis, work performed after the original PRA, and discovery of some errors, it was found to be necessary to re-evaluate certain scenarios before using them to compare alternatives. As a result of this re-evaluation, discussed in the previous section, the estimates of certain accident frequencies examined here went down very considerably. For present purposes, only selected scenarios were re-evaluated; it is difficult to say whether all other PRA scenarios would change as a result of a comparable re-evaluation.

The accidents chosen as a representative basis for comparing alternatives are shown in Table C.5-14. The current approach is to characterize each of these accidents for each alternative to which the accident is applicable.

Although quantified at a high frequency, BTR was found to be subject to significant conservatism and/or significant modeling uncertainty, and therefore unsuitable as a basis for comparing alternatives. LOOP is equivalent to a large class of accidents in its potential consequences and exhibits a meaningful variation of those consequences with operating power level, and is potentially less subject to conservatism and uncertainty than BTR. LOOP is therefore a potentially useful accident for comparison of alternatives. However, a reassessment of its progression reduced the estimate of its frequency to a level at which it is doubtful that LOOP alone can be considered a properly representative contributor. Large LOCA was therefore examined as a possible supplement or replacement. Very significant uncertainties are associated with large LOCA, but it has been selected for use in the comparison. A case could be made for comparing on the basis of ATWS scenarios. At this point, it seems that there exists sufficient uncertainty regarding the frequency and accident initiation of ATWS scenarios to tend to disqualify them as a basis for comparison of alternatives.

***Table C.5-14. Representative Accidents***

| Accident Sequence Initiator | Relative Magnitude of Consequences of Sequence Analyzed | Comments |
|---|---|---|
| LOOP | Moderate | Low assessed frequency; consequences typical of major core damage accidents at this facility; frequency and consequence magnitude vary somewhat with power level |
| Large LOCA | Low (60 MW), None (30 | In the scenario analyzed, minor core damage |

| | | |
|---|---|---|
| | MW) | occurs at 60 MW, but the core is subsequently cooled. No core damage at 30 MW. |
| SWT | High | Low-frequency scenario, but a significant contributor to frequency-weighted offsite consequences; consequence magnitude varies somewhat with power level |
| FHA | Low | Most severe within-design-basis event. Consequences vary minimally with power level. Frequency varies because the number of elements handled per year is higher at 60 MW than at 30 MW. |
| $D_2O$ Release | Very Low | No airborne release, but unmitigated event might contaminate groundwater |
| Experimental Facilities Release | Very Low | Confinement function should limit consequences of such events to very low levels |

In this connection, it is worth re-emphasizing that almost all major core damage sequences are assessed to have roughly the same consequence magnitudes, as long as confinement is not breached. If not for the need to compare the characteristics of specific accidents across different alternatives, one would try to derive a characteristic overall core damage frequency, and associate with that frequency the typical consequence magnitude.

SWT is one of the biggest contributors to frequency-weighted consequences; this scenario has an originally-estimated frequency that is high compared to other scenarios leading to “major” core damage and, as a result of a breached confinement, very nearly the largest radiological release. The frequency was conservatively modeled originally, and recent facility modifications serve to reduce the frequency even if the conservative intent was maintained in the modeling. While a limited reanalysis has shown the CDF for tornadoes to be significantly reduced, the large radiological release associated with the failed confinement makes this accident of interest for comparison of alternatives.

The postulated FHA was analyzed both in the PRA and in the SAR. The PRA presented the frequency, but not the consequences; the SAR did not explicitly analyze the frequency of the event, and presented conservatively computed dose consequences to the MEI but not to any population. Therefore, in order to generate the information needed for present purposes, MACCS analyses were performed for this event (see Section C.5.1.1.3). The SAR approach to consequence analysis is more conservative than is the MACCS approach, and results for the MEI should therefore not be compared between the two calculations.

The last two postulated accidents, $D_2O$ release and releases from experimental facilities, have relatively low consequences per event but potentially observable frequencies. As will be seen, the $D_2O$ release also distinguishes non-operating alternatives from each other.

## C.5.2 NO ACTION ALTERNATIVE

### C.5.2.1 Accident Scenarios and Source Terms

None of the accidents in the portfolio of representative accidents comes into play in the No Action Alternative in the same form in which they were defined. For instance, the core damage accidents cannot occur since there is no fuel in the facility.

Scenarios that are functionally equivalent to the two lowest-consequence scenarios ($D_2O$ spill and release from experimental facilities) could occur during the conduct of maintenance or modification activities. It is possible to have a spill of $D_2O$, but only so long as the $D_2O$ is kept within the HFBR facilities; it is possible to have a fire in the HFBR with the confinement structure open, driving small amounts of radioactive material into the air, but only so long as the material is left in place; and so on. Based on the above, a $D_2O$ release is used as the representative accident for this alternative. The source term for this release is assumed to be equivalent to the source term used for the operational $D_2O$ release scenario discussed in Section C.5.1.1.2.6.

A comparable assumption has not been made for a release from experimental facilities. Under the No Action Alternative, it is doubtful that experimental apparatus would be admitted to the HFBR.

### C.5.2.2 Accident Impacts

The radiological health consequences are minimal under the stated assumptions. An involved worker would receive a dose of approximately 1 mrem from the $D_2O$ accident (BNL 1998), which would result in a probability of LCFs of $4x10^{-7}$ (or a 4 in 10,000,000 chance) per accident for the worker. Because the impact to the involved worker is so low, further analysis to determine the impact to noninvolved workers and the public was not performed.

## C.5.3 RESUME OPERATION ALTERNATIVE – 30 MW POWER LEVEL

### C.5.3.1 Accident Scenarios and Source Terms

All of the scenarios in the representative potential accident set pertain to this alternative and are addressed in Table C.5-15. Variation in accident frequency for LOOP and SWT with power level does not reflect a different assumption regarding the LOOP or the SWT initiating events; it reflects some difference in the fraction of time during which the accident is possible, owing to the reactor operating for different fractions of the time at different power levels. It also reflects different times available for operator action owing to the differences in the physics of accident progression for accidents initiated at different operating power levels. The LOCA greater than 33 cm (13 in) with no other failures and no core damage is shown for purposes of comparison with 60 MW, because at 60 MW, the same event has different consequences.

### C.5.3.2 Accident Impacts

The accident impacts pertaining to this alternative are summarized in Table C.5-15. This table indicates that the consequences of the SWT accident (which is a beyond design basis accident) are worse than those of the other accidents if the SWT accident actually occurs as postulated, but the frequency of the SWT accident is low, and the risk (as measured by possible frequency multiplied by the possible consequences) posed by this accident is relatively minor. The consequences and risks of the other scenarios are all less than the consequences and risk of the SWT accident. For consequences for LOCA and FHA at 30 MW are shown to be extremely small (less than 0.1 LCF to the public).

In the case of the LOOP accident, the dose to the noninvolved onsite worker actually seems to be worse at 30 MW. This arises because the consequences to the noninvolved onsite worker do not simply reflect release characteristics, but also depend on assumptions regarding evacuation of onsite workers, which is assumed to be determined by developments in the accident sequence.

# C.5.4 RESUME OPERATION ALTERNATIVE – 60 MW POWER LEVEL

## C.5.4.1 Accident Scenarios and Source Terms

All of the scenarios in the representative set of accidents can occur in this alternative, and are discussed as such in Section C.5.1.1.2. Variation in accident frequency for LOOP and SWT with power level does not reflect a different assumption regarding the LOOP or the SWT initiating events; it reflects some difference in the fraction of time during which the accident is possible, owing to the reactor operating for different fractions of the time at different power levels. It also reflects different times available for operator action owing to the differences in the physics of accident progression for accidents initiated at different operating power levels. The LOCA greater than 33 cm (13 in) causes minor core damage at 60 MW but not at 30 MW. The consequences at 60 MW are relatively minor because the EFC then stabilizes the core.

## C.5.4.2 Accident Impacts

The accident impacts pertaining to this alternative are summarized in Table C.5-16. This table indicates that the consequences of the SWT accident (which is a beyond design basis accident) are worse than those of the other accidents if the SWT accident actually occurs as postulated, but the frequency of the SWT accident is low, and the risk (as measured by frequency multiplied by the consequences) posed by this accident is relatively minor. The consequences and risks of the other scenarios are all less than the consequences and risk of the SWT accident. In comparison to the offsite accident consequences for 30 MW operation, the offsite consequences of the SWT and LOOP accidents (which are both beyond design basis accidents) at 60 MW operation are seen to be about 50 percent greater than for 30 MW operation. For consequences for LOCA and FHA at 60 MW are shown to be extremely small (less than 0.1 LCF to the public).

Note that 97 percent of the noninvolved workers are assumed to evacuate offsite following the LOOP accident at both 30 MW and 60 MW, and that these evacuated noninvolved workers receive minimal doses in comparison to the noninvolved workers who remain onsite. Onsite relocation is assumed to occur based on exceeding a projected dose rate limit (Wagage 1999). For the LOOP accident, because the accident dose rate is greater for 60 MW than at 30 MW operations, more noninvolved workers would be relocated at 60 MW than at 30 MW. For this accident, the reduced doses received by the extra noninvolved workers that relocate at 60 MW more than offset the increased doses received by the non-relocated, noninvolved workers at 60 MW. The net result is that the noninvolved worker population for 30 MW operation is calculated to receive a population dose per accident that is two person-rem (less than one percent) greater than the population dose per accident for 60 MW operation.

Doses to experimenters and other facility workers from design basis accidents (other than facility operators who are responding to the emergency) would be minimal. Doses to experimenters and other facility workers from beyond design basis accidents have not been systematically assessed. Most of the postulated accidents leading to core damage would proceed slowly enough that experimenters and other facility workers would leave the facility before dose rates could become significant. one possible exception to this is the postulated LOCA large enough to lead to minor core damage at 60 MW but not at 30 MW. (This LOCA has a frequency conservatively estimated at $6x10^{-5}$ per year.) Because of the minor core damage, doses to operators responding to this event at 60 MW were estimated to be 2.6 rem (see BNL 1993 and Section C.5.1.1.2.1). This dose, while well above doses from normal operations, would not itself exceed annual occupational dose limits. The released coolant itself would pose a separate (but lesser) hazard. Pending a systematic assessment of consequences to experimenters and other facility workers, this 2.6 rem dose is taken to bound the consequences to experimenters and other facility workers from the large LOCA at 60 MW. The other accidents discussed here would not exhibit a significant difference in doses to experimenters and other facility workers at 30 MW and 60 MW.

## C.5.5 RESUME OPERATION AND ENHANCE FACILITY ALTERNATIVE

### C.5.5.1 Accident Scenarios and Source Terms

Based on existing work, the Resume Operation and Enhance Facility Alternative does not change any of the accident scenarios in the representative set from their values at 60 MW Resume Operation Alternative.

As explained in Section C.5.1.1.2.3, it is arguable whether the Resume Operations and Enhance Facility Alternative would change the probability assessed for the BTR accident, which was not chosen as a representative accident for purposes of the present discussion. In the PRA, BTR was an extremely important contributor to assessed CDF. It was also a highly uncertain contributor, and its importance was driven by the uncertainty that was associated with it. The uncertainties associated with BTR are summarized above in Section C.5.1.1.2.3.

If the BTR frequency were actually considered to represent our state of knowledge, and if revesseling indeed improved the components significantly, then it would be useful to reflect this in a comparison of the Resume Operation and Enhance Facility Alternative with other operating alternatives. If there were evidence that revesseling would affect the assessed frequency of BTR, then it would be important to point this out, even if the current frequency estimate were debatable. However, this is not the case. The present reading of the PRA is that the quantification of BTR frequency is not driven by evidence of current problems in materials behavior, but rather is driven simply by uncertainty. It is therefore not clear that revesseling would change the BTR frequency.

### C.5.5.2 Accident Impacts

The accident impacts pertaining to this alternative are the same as are given in Table C.5-16 and are discussed in Section C.5.4.2.

## C.5.6 PERMANENT SHUTDOWN ALTERNATIVE

### C.5.6.1 Accident Scenarios and Source Terms

None of the potential accidents in the portfolio of representative accidents comes into play in the Permanent Shutdown Alternative in the same form in which they were defined. For instance, the core damage accidents cannot occur since there is no fuel in the facility.

Scenarios that are functionally equivalent to the two lowest-consequence scenarios ($D_2O$ release and release from experimental facilities) could occur during a transition to a permanent shutdown state, but not once such a transition had been made. It is possible to have a release of $D_2O$, but only so long as the $D_2O$ is kept within the HFBR facilities; it is also possible to have a fire in the HFBR with the confinement structure open, driving small amounts of radioactive material into the air, but only so long as the material is left in place; and so on. Based on the above, a $D_2O$ release bounds any possible accident, and so is used as the representative accident for this alternative.

### C.5.6.2 Accident Impacts

The accident impacts pertaining to this alternative are the same as are discussed in Section C.5.2.2.

### *Table C.5-15. 30 MW Operation Alternative Accident Impacts at the HFBR*

| | Onsite Noninvolved Worker Population | | Maximally Exposed Offsite Individual | | Population to 80 km | | |
|---|---|---|---|---|---|---|---|
| **Accident Description** | **Population Dose Per Accident[d] (person-rem)** | **Number of LCFs Per Accident** | **Dose Per Accident (rem)** | **Probability of LCF** | **Population Dose Per Accident[e] (person-rem)** | **Number of LCFs Per Accident** | **Accident Frequency (per year)** |
| LOOP[a] | 288 | 0.12 | 0.64 | $3x10^{-4}$ | 8,400 | 4.2 | $8.6x10^{-8}$ [f] |
| Large LOCA[b] | None [b] | None [b] | None [b] | None [b] | None [b] | None [b] | $6.5\ x10^{-5}$ |
| SWT[c] | 2,900 | 1.1 | 61 | $6x10^{-2}$ | 160,000 | 81 | $7.9x10^{-7}$ [f] |
| FHA | 4 | 0.0016 | 0.0077 | $4x10^{-6}$ | 59 | 0.03 | $2.6x10^{-5}$ |

[a] Normal cooling function not available, core water inventory not replenished; core damage occurs. Ex-confinement release is somewhat filtered.

[b] Event postulated is a large break (greater than 33 cm) successfully cooled at 30 MW with no core damage. Event is postulated for comparison with 60 MW, at which minor core damage occurs for a break of this size.

[c] Severe wind/tornado causes loss of offsite power, breaches confinement with projectile and also eliminates coolant makeup. Ex-confinement release not filtered because confinement is breached.

[d] Based on a total non-involved worker population of 2,686.

[e] Based on a total offsite population of 5,356,270. This population and the associated population distribution were calculated using SECPOP90 (Humphreys 1997). SECPOP90 is a computer program that provides population and economic data estimates for any location in the U.S. with the results available in MACCS site file format. MACCS is the code used to calculate accident radiological consequences (SNL 1990a, SNL 1990b, SNL 1990c). Note that, because of the differences in population input files for the MACCS code and the CAP88-PC model (the code used to calculate radiological consequences from normal operations, see EPA 1992,), a different offsite population (5,053,187) was used to calculate offsite doses from normal operations. The population input file for the CAP88-PC model was derived from customer records of LILCO (now LIPA).

[f] The LOOP and SWT accident frequencies reflect not only the frequency of initiating events, which would be the same at both power levels, but also subsequent failures, whose probabilities differ at different power levels because different times are available for actions to be taken. See Sections C.5.1.1.2.2 and C.5.1.1.2.4, respectively.

Notes:

1. The frequency of the FHA is obtained by scaling the PRA result for 60 MW by the relative number of fuel elements handled at 30 MW.
2. The consequence estimates presented here for LOOP, Large LOCA, SWT, and FHA scenarios are based on calculations discussed in C.5.1.1.3.
3. The frequency of breaks greater than 33 cm is estimated based on arguments given in BNL 1990, BNL 1990b.
4. A $D_2O$ release accident was postulated but was not evaluated in detail, and is not shown, because the involved worker would receive approximately 1 mrem from this accident, and noninvolved workers and the public would receive much less.
5. An Experimental Facility Accident comparable to the TRISTAN fire was postulated but was not evaluated in detail because its consequences were negligible.

Source:  BNL 1990a, BNL 1990b, BNL 1993b, BNL 1998, Schmidt 1998, Wagage 1999, Palmrose 1999.

### *Table C.5-16. 60 MW Operation Alternative Accident Impacts at the HFBR*

| | Onsite Noninvolved Worker Population | | Maximally Exposed Offsite Individual | | Population to 80 km | | |
|---|---|---|---|---|---|---|---|
| **Accident Description** | **Population Dose Per Accident[d] (person-rem)** | **Number of Latent Cancer Fatalities Per Accident** | **Dose Per Accident (rem)** | **Probability of Latent Cancer Fatality** | **Population Dose Per Accident[e] (person-rem)** | **Number of Latent Cancer Fatalities Per Accident** | **Accident Frequency (per year)** |
| LOOP[a] | 286 | 0.11 | 1.1 | $6x10^{-4}$ | 12,000 | 6.2 | $2.6x10^{-7}$ [f] |
| Large LOCA[b] | 11 | 0.0046 | 0.022 | $1x10^{-5}$ | 149 | 0.075 | $6.5x10^{-5}$ |
| SWT[c] | 3,300 | 1.3 | 110 | 0.11 | 230,000 | 115 | $8.7x10^{-7}$ [f] |
| FHA | 4.6 | 0.0018 | 0.0082 | $4x10^{-6}$ | 68 | 0.03 | $6.0x10^{-5}$ |

[a] Exclusive of ATWS. Normal cooling function not available, core water inventory not replenished; core damage occurs. Ex-confinement release is somewhat filtered.
[b] Event postulated is a large break (greater than 33 cm) with minor core damage, stabilized thereafter by EFC.
[c] SWT causes LOOP, breaches confinement with projectile and also eliminates coolant makeup. Ex-confinement release not filtered because confinement is breached.
[d] Based on a total non-involved worker population of 2,686.
[e] Based on a total offsite population of 5,356,270. This population and the associated population distribution were calculated using SECPOP90 (Humphreys 1997). SECPOP90 is a computer program that provides population and economic data estimates for any location in the U.S. with the results available in MACCS site file format. MACCS is the code used to calculate accident radiological consequences (SNL 1990a, SNL 1990b, SNL 1990c). Note that, because of the differences in population input files for the MACCS code and the CAP88-PC model (the code used to calculate radiological consequences from normal operations, see EPA 1992,), a different offsite population (5,053,187) was used to calculate offsite doses from normal operations. The population input file for the CAP88-PC model was derived from customer records of LILCO (now LIPA).
[f] The LOOP and SWT accident frequencies reflect not only the frequency of initiating events, which would be the same at both power levels, but also subsequent failures, whose probabilities differ at different power levels because different times are available for actions to be taken. See Sections C.5.1.1.2.2 and C.5.1.1.2.4, respectively.

Notes:
1. The frequency of the FHA is obtained from the PRA (Table C.5.1.1.1-2).
2. The consequence estimates presented here for LOOP, Large LOCA, SWT, and FHA scenarios are based on calculations discussed in C.5.1.1.3.
3. The frequency of breaks greater than 33 cm is estimated based on arguments given in BNL 1990, BNL 1990b.
4. A $D_2O$ release accident was postulated but was not evaluated in detail, and is not shown, because the involved worker would receive approximately 1 mrem from this accident, and noninvolved workers and the public would receive much less.
5. An Experimental Facility Accident comparable to the TRISTAN fire was postulated but was not evaluated in detail because its consequences were negligible.

Source:  BNL 1993b, BNL 1998, Schmidt 1998, Wagage 1999, Palmrose 1999.

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